6

PNL-4673

Fundamentals of Health Physics for the Radiation Protection Officer

March 1983

Prepared for the
U.S. Department of the Army

Pacific Northwest Laboratory
Operated for the U.S. Department of Energy
by Battelle Memorial Institute

Battelle

PACIFIC NORTHWEST LABORATORY
operated by
BATTELLE
for the
UNITED STATES DEPARTMENT OF ENERGY
under Contract DE-AC06-76RLO 1830

PNL-4673

FUNDAMENTALS OF HEALTH PHYSICS FOR THE RADIATION PROTECTION OFFICER

B. L. Murphy
R. J. Traub
R. L. Gilchrist
J. C. Mann
L. H. Munson
E. H. Carbaugh
J. L. Baer

Contributors:
L. W. Brackenbush
T. H. Essig
D. E. Hadlock
W. N. Herrington
M. P. Moeller
D. W. Murphy
J. M. Selby
J. E. Tanner
J. M. Taylor
C. M. Unruh

March 1983

Prepared for
the U.S. Department of the Army
under a Related Services Agreement
with the U.S. Department of Energy
Contract DE-AC06-76RLO 1830

Pacific Northwest Laboratory
Richland, Washington 99352

ACKNOWLEDGMENTS

The authors wish to express their very great appreciation to Marianna Cross for the preparation of this manuscript for publication. In addition, the comments and assistance received from staff members of the Radiological Sciences Department are gratefully acknowledged.

CONTENTS

CHAPTER 1. PROPERTIES OF RADIOACTIVE MATERIALS

FIGURES

TABLES

CHAPTER 1. PROPERTIES OF RADIOACTIVE MATERIALS

The world around us is composed of elements and combinations of elements, each with its own unique chemical properties. Only about 100 elements are known to man. Some examples are hydrogen, oxygen, carbon, gold, and silver. Substances such as water, wood, rock, rubber, coal, and hundreds of thousands more are combinations of the comparatively few elements. These combinations are called compounds.

Each element can be denoted by a one- or two-letter chemical symbol; for example, H is the symbol for hydrogen, O is the symbol for oxygen, and Au is the symbol for gold. Compounds are denoted by combinations of element symbols and numbers that refer to the proportion of each element in the compound. Water, for example, which has two units of hydrogen for every unit of oxygen, is designated H_2O. A list of all of the known elements and the chemical symbol for each can be found in Chapter 16, "Reference Data."

Some atoms are unstable and undergo transitions that result in the formation of a more stable atom and the release of some energy. This process is called radioactive decay, and substances that are unstable and subject to decay are called radioactive materials.

This chapter provides a review of the fundamental characteristics of radioactivity. The initial portion covers basic information about atomic structure and radioactive decay. The properties of ionizing radiation are then reviewed, followed by a discussion of radiation quantities and units. Information on the biological effects of radiation is presented. The chapter concludes with the presentation of concepts important to the development of radiation protection procedures.

Section 1.1 ATOMIC STRUCTURE

The smallest unit of an element is the atom. An atom consists of a small, dense, positively charged nucleus surrounded by a cloud of negatively charged electrons.

1.1.1 The Nucleus

The nucleus consists of two fundamental particles, protons and neutrons. The proton is a positively charged particle that has a unit charge of 1.6×10^{-19} coulombs. The mass of a proton is 1.67×10^{-24} gram. The number of protons in the nucleus, the atomic number, Z, is unique for each element; for example, if a nucleus contains six protons, the atom is a carbon atom; on the other hand, if a nucleus contains eight protons, the atom is an oxygen atom.

The neutron is a particle that has no electrical charge and has a mass slightly greater than that of a proton. The nuclei of the atoms that make up a given element may contain varying numbers of neutrons. The number of neutrons in the nucleus, the neutron number, N, influences the stability of the nucleus; that is, it determines whether the atom is radioactive. If the N number of a nucleus is plotted as a function of the Z number of the nucleus, as shown in Figure 1.1, stable, or nonradioactive, nuclei tend to be clustered about a line called the line of stability. In the case of nuclei of low Z, the most stable nuclei have approximately equal numbers of protons and neutrons. In the case of very heavy nuclei (those with many protons, or high Z), the nucleus is most stable if the number of neutrons in the nucleus is about 1.5 times the number of protons.

Isotopes are atoms of one element that have the same atomic number but that differ in neutron number. The isotopes of a given element have the same chemical properties and cannot be separated by chemical methods. However, the nuclear characteristics of the isotopes may be quite different; for example, some isotopes of an element may be radioactive while others are not. Isotopes of a given element are identified by their mass number, A, which is the total number of protons plus neutrons in the nucleus; that is, $A = Z + N$.

Individual atoms are called nuclides; the radioactive forms are called radionuclides. An isotope or nuclide may be identified by its chemical symbol, with the atomic number, Z, as a presubscript and the mass number, A, as a presuperscript: ${}^{A}_{Z}X$, where X represents the chemical symbol. Because the atomic number, Z, is unique to a given element, it is often omitted from this notation. Sometimes a nuclide is designated by the full name of the element,

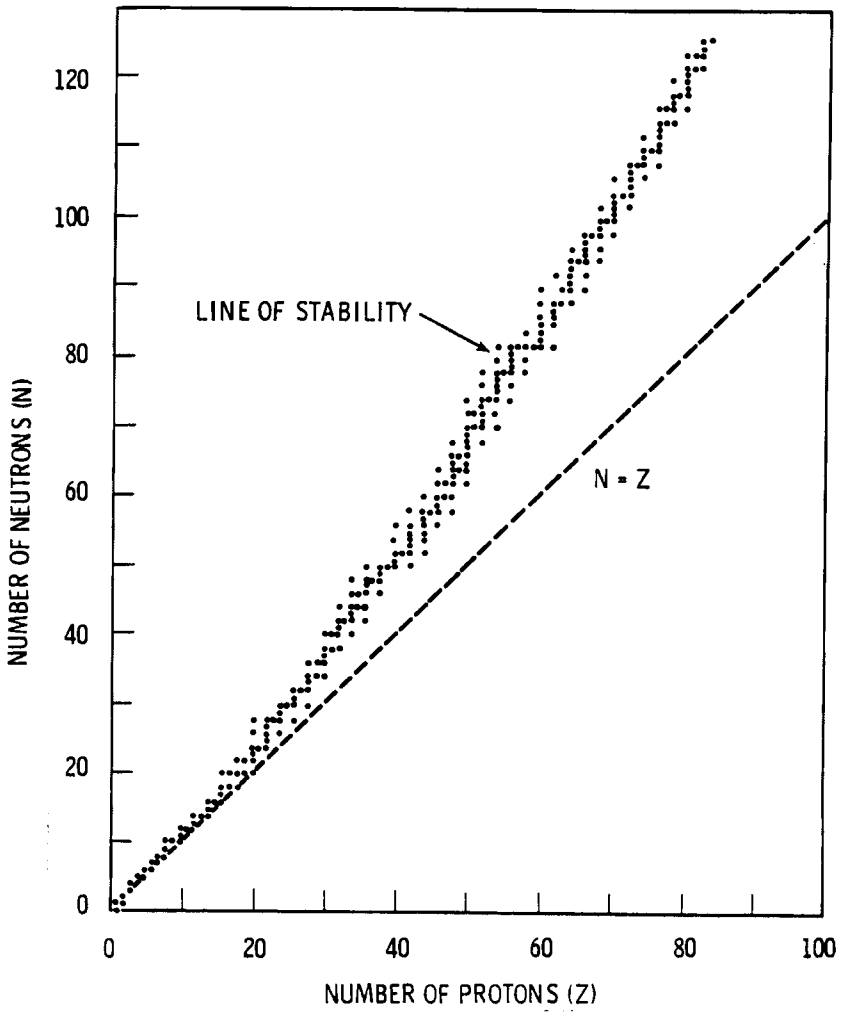


FIGURE 1.1. Numbers of Neutrons and Protons in Stable Nuclides

or its chemical symbol, followed by a hyphen and the A number. Thus, ^{12}C, $^{12}_{6}C$, C-12, and carbon-12 are four ways of designating the same nuclide. In the past, the A number was written with the chemical symbol as a postsuperscript, C^{12}.

The natural elements of the earth's crust or atmosphere are composed of mixtures of the isotopes of each element. The isotopes vary in their percent natural abundance; that is, they do not all occur in equal amounts. For example, of all the oxygen atoms that occur on earth, 99.756% are ^{16}O, 0.034%

are ^{17}O, and 0.205% are ^{18}O. The relative abundance of stable isotopes remains fairly constant over a wide geographic range.

1.1.2 Electrons

The nucleus is surrounded by electrons, which have a negative charge that is equal in magnitude, but opposite in sign, to that of the proton. In the neutral, uncharged atom, there is one electron outside the nucleus for every proton in the nucleus. The electrons can be thought of as occupying orbits, or shells, as shown in Figure 1.2. Because the photons give the nucleus a positive charge and the electrons have a negative charge, and because opposite charges tend to attract each other, there is an attractive force between an atom's nucleus and its electrons. The shells represent the strength of the attractive force between the nucleus and the electrons, not the exact location of the electrons.

The shells form a series of energy or quantum levels. The diameters of the shells are large in comparison with the diameter of the nucleus. The

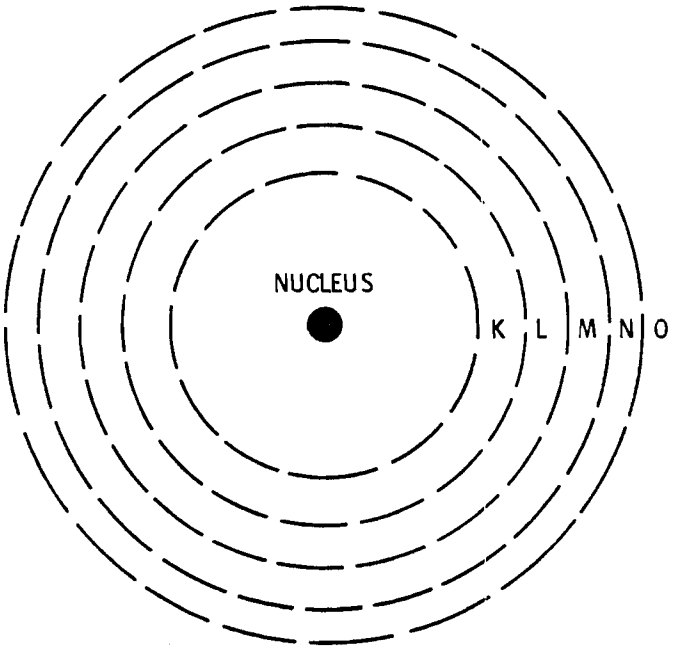


FIGURE 1.2. Schematic Diagram of an Atom Showing Nucleus and Electron Shells

shells are identified by either a letter (K, L, M, N, O, P, Q) or a quantum number (1, 2, 3, 4, 5, 6, 7). The energy state of each electron in a shell is completely described by four independent quantum letters (n, l, m, and s), and the Pauli Exclusion Principle sets an upper limit on the number of possible electrons in each shell.

Because of the attractive force between the nucleus and the electrons, it takes a certain amount of energy to remove the electrons from the atom. The amount of energy required to completely remove an electron from the atom is called the electron binding energy. This energy is different for each shell in the atom of any one element, and different for the same shell in different elements. The electrons closest to the nucleus, in the K shell, have a greater attraction to the nucleus than electrons farther from the nucleus. The electron binding energy associated with an inner shell is therefore greater than that of an outer shell.

If an electron is removed from an inner shell, a vacancy, or "hole," is formed in that shell. An electron from one of the outer shells may then "jump" or "fall" into the vacancy. When this happens, energy equal to the difference between the electron binding energies of the two shells is emitted from the atom in the form of electromagnetic radiation. This radiation is called characteristic radiation because the amount of energy released is characteristic of a given element. Characteristic radiation may be given off in the form of light, heat, or x rays, depending upon its energy.

Section 1.2 RADIOACTIVITY AND RADIOACTIVE DECAY

Radioactivity is the tendency of unstable nuclides to undergo radioactive decay. Radioactive decay is defined as a spontaneous, energy-releasing atomic transition that involves a change in the state of the nucleus of an atom. This change means that the atom changes from one nuclide (the parent) into a second nuclide (the daughter) or from one nuclear energy level to a lower energy level. The difference in the energy levels determines the amount of energy released by the transition. The transition must be spontaneous, that is, free from the influence of outside forces. It is possible to use machines

such as cyclotrons, linear accelerators, or even nuclear reactors to change the nucleus of an atom; however, such transitions are not considered radioactive decay.

1.2.1 Characterization of Radionuclides

A radioactive nuclide, or radionuclide, can be characterized by its rate of decay, the energy released during the decay, and the type of radiation emitted by the decay.

A. Rate of Decay. All radionuclides do not decay at the same rate. Some decay very quickly, in a matter of a few seconds. Others may take days, weeks, or millions of years to decay. The rate of decay of a radionuclide is measured in terms of a half-life.

The half-life of a radionuclide, symbolized $t_{1/2}$, is the time required for the number of radioactive atoms present to decrease by one half. After one half-life, 50% of the original radioactive atoms remain; after two half-lives, 25% of the original radioactive atoms remain; etc. Figure 1.3 illustrates the concept of half-life using ^{198}Au, an isotope of gold, as an example. The half-life of a particular radionuclide may be found in the Table of Isotopes (Lederer and Shirley 1978) or the Radiological Health Handbook (1970).

The rate of radioactive decay can also be expressed in terms of the decay constant, λ, of the radionuclide. The decay constant indicates the fraction of radioactive atoms present that will undergo radioactive decay in a given period of time. It is numerically equal to the natural logarithm of 2 (0.693) divided by the half-life of the radionuclide. That is,

$$\lambda = (\ln 2)/t_{1/2} = 0.693/t_{1/2} \qquad (1.1)$$

The decay constant is used when calculating the number of radioactive atoms present in a sample at any time, using the equation

$$N = N_0 e^{-\lambda t} \qquad (1.2)$$

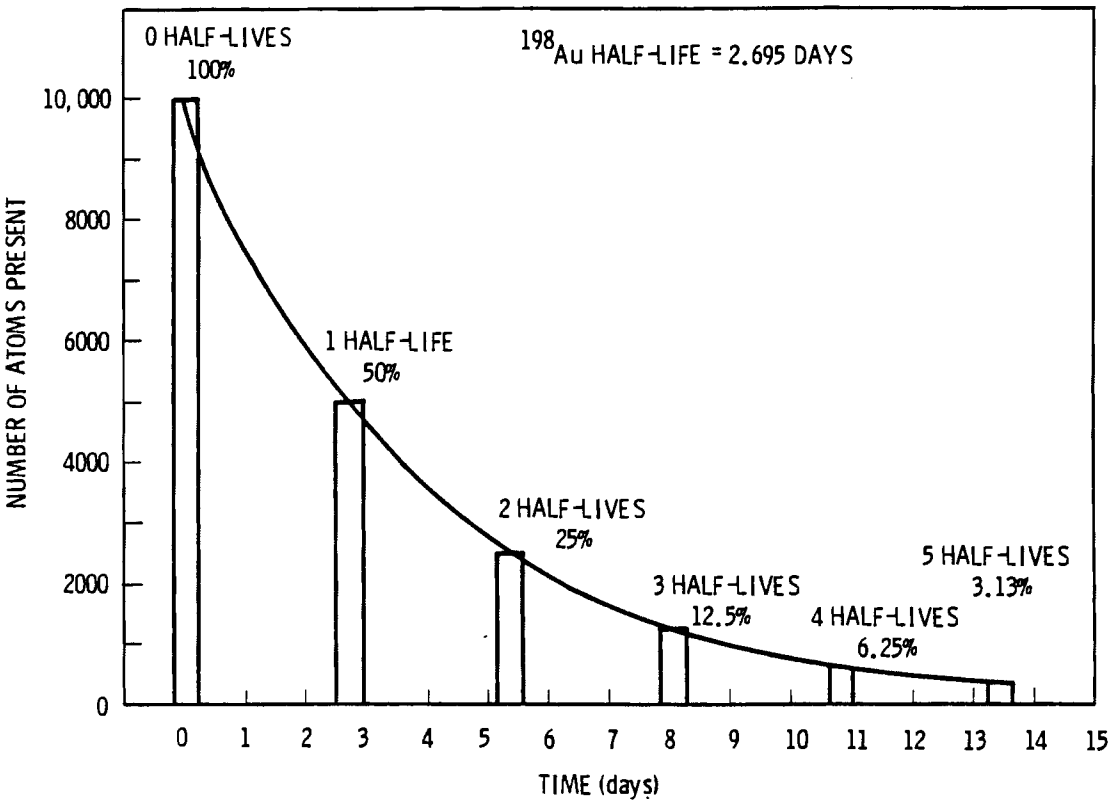


FIGURE 1.3. Number of ^{198}Au Atoms Present as a Function of Half-Lives Elapsed

where N = the number of radioactive atoms present at time t

N_o = the number of radioactive atoms originally present

e = the base of the natural logarithms (2.71828)

λ = the decay constant of the given radionuclide

$= (\ln 2)/t_{1/2} = 0.693/t_{1/2}$

t = the time elapsed.

The half-life and decay constant are inversely related. A radionuclide with a long half-life has a small decay constant; a radionuclide with a short half-life has a relatively large decay constant.

B. Energy of Decay. The unit of energy used in radioactive decay is the electron volt (eV). The electron volt, which is the energy acquired by an electron when it falls through a potential difference of 1 volt, is equal to

about 1.6×10^{-19} joules. Multiples of the electron volt, such as the kilo-electron volt (keV) and the millionelectron volt (MeV), are also used. One keV equals 1000 eV and 1 MeV equals 1,000,000 eV.

The energy of radioactive decay is observed as the kinetic energy of particulate radiation, electromagnetic radiation, or as both.

(1) Kinetic Energy of Particles. Radioactive decay can change the state of an atom's nucleus through the emission of particles from the nucleus. These particles have kinetic energy, or energy of motion. The kinetic energy (T) of a particle is a function of its mass (m) and its velocity (v). According to classical physics,

$$T = \frac{1}{2} mv^2 \qquad (1.3)$$

From this equation we learn that, if two particles have the same velocity, their kinetic energies are related by a simple ratio of their masses. Conversely, two particles of equal kinetic energy have velocities that are related to the square root of their masses. That is, a light particle has a velocity greater than that of a heavy particle of equal kinetic energy.

Equation (1.3) is valid if the velocity of the particle is not comparable to the velocity of light. When the speed of the particle becomes faster than one-tenth the speed of light, the mass of the particle increases, and the equation cannot be used. Particles traveling at velocities comparable to the speed of light are said to be traveling at relativistic velocities, and the equation relating the kinetic energy of a particle and its velocity is then

$$T = m_o c^2 \left[\frac{1}{\sqrt{1 - \beta^2}} - 1 \right] \qquad (1.4)$$

where T = the kinetic energy of the particle
m_o = the mass of the particle
c = the velocity of light
$\beta = v/c$
v = the velocity of the particle.

An important consequence of this equation is that no particle, whatever its energy, can travel faster than the speed of light in a vacuum. Figure 1.4 illustrates the energy-velocity relationship for alpha and beta particles.

(2) Electromagnetic Energy. The energy released from a decaying radionuclide can also take the form of oscillating (vibrating) electric and magnetic fields, or electromagnetic radiation. This radiation travels in the form of waves that have a characteristic frequency, υ, and wavelength, λ. The frequency of electromagnetic radiation is expressed in terms of cycles per second, or Hertz (Hz). The wavelengths of various electromagnetic radiations are expressed in units of measure appropriate to their length. For example, wavelengths of ultraviolet radiation are measured in nonometers or meters, whereas radio waves are measured in centimeters or meters. All electromagnetic radiations travel at the velocity of light, which is about 3×10^8 meters per second (m/sec) in a vacuum. The wavelength times the frequency is equal to the the velocity of light. The electromagnetic wave spectrum consists of wavelengths ranging from

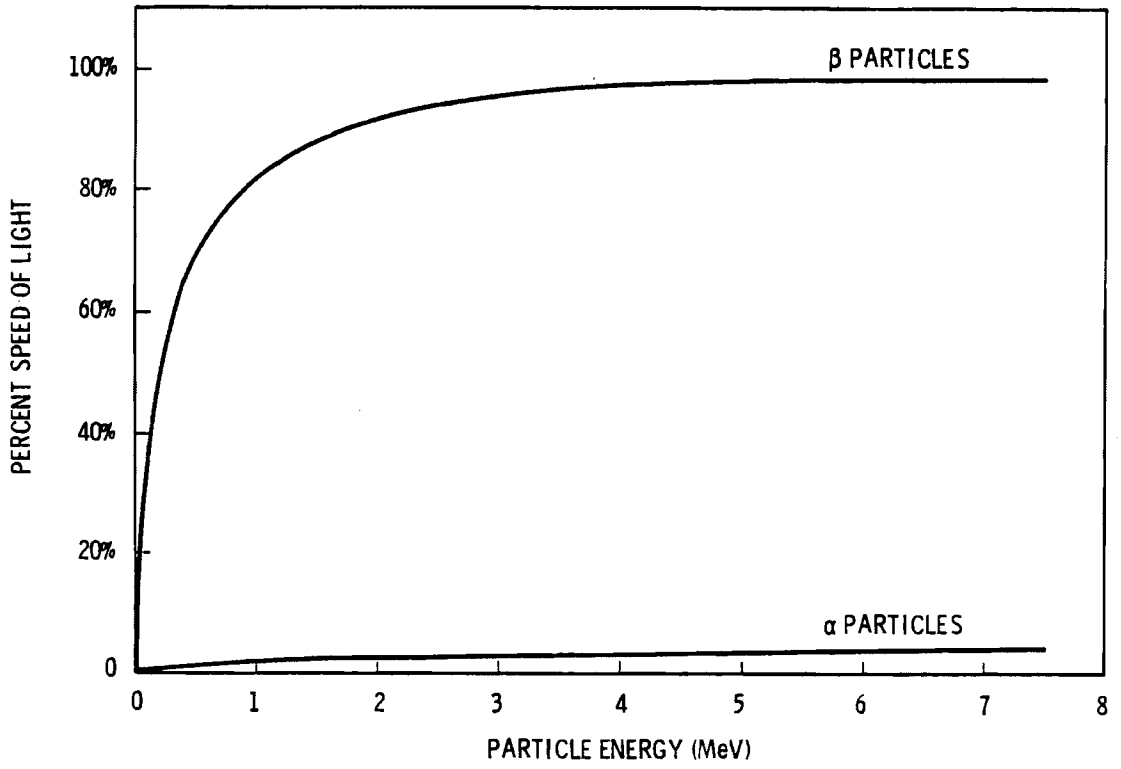


FIGURE 1.4. Energy-Velocity Relationships for Alpha and Beta Particles

several kilometers to a small fraction of a nanometer (10^{-9} m). Between these limits lies a continuous range of electromagnetic waves.

Figure 1.5 shows the electromagnetic spectrum. This spectrum is divided into a number of regions, each representing wavelength intervals within which there is a common state-of-the-art in radiation sources and detectors. All of these regions overlap; that is, the characteristics of the radiation change slowly with the change in frequency, and it is difficult to know exactly where one region ends and the next begins. Examples of electromagnetic radiation include radio waves and microwaves, infrared and visible light, and x and gamma radiation.

Electromagnetic radiation exists as waves; however, when discussing the energy of electromagnetic radiation, it is often convenient to think of the

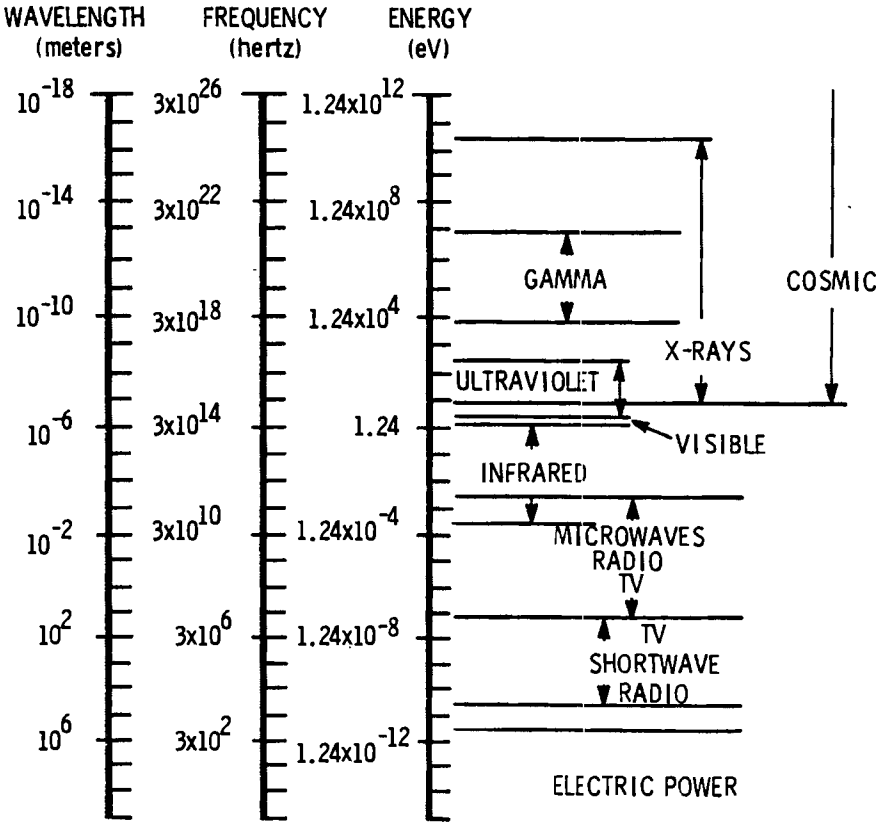


FIGURE 1.5. The Electromagnetic Spectrum

waves as existing in the form of wave packets, called quanta or photons. At high energies, these wave packets or photons behave as if they were small particles. This phenomenon of electromagnetic radiation acting like particles or particles acting like waves is called the wave-particle duality of electromagnetic radiation.

The energy of a single photon or quantum is related to the frequency of the radiation and ranges from very small values at low frequencies to very large values at high frequencies. That is,

$$E = h\upsilon \tag{1.5}$$

where E = the energy associated with a photon of electromagnetic radiation

h = Planck's constant (4.136×10^{-15} eV·sec)

υ = the frequency.

C. Type of Radiation Emitted. When particulate or electromagnetic radiation has an energy greater than about 30 eV, it is able to strip an electron from a molecule in a process called ionization. Photon energy is sufficient to cause ionization at frequencies greater than that of light. Radiation with this high an energy level is called ionizing radiation. The most important characteristic of ionizing radiation is its localized release of large amounts of energy, approximately 33 eV per ionizing event. This amount of energy is more than enough to break a strong chemical bond; for example, the energy associated with a C=C bond, commonly found in body tissues, is 4.9 eV. The ability to break chemical bonds makes ionizing radiation of concern because it can disrupt the function of living cells.

Radioactive decay results in five types of ionizing radiation: alpha particles, beta particles, gamma rays, x rays, and neutrons. These types of radiation can be distinguished by their physical characteristics, such as mass, electrical charge, and path length or range, as shown in Table 1.1. The two major modes of decay result in the emission of alpha particles or beta particles. Both of these decay modes can also be accompanied by the emission of gamma rays. The five types of ionizing radiation and the types of decay that produce them are described below.

TABLE 1.1. Radiation Characteristics

Radiation	Mass (g)	Charge[a]	Path Length	
			Air	Solid
Alpha particles	6.64×10^{-24}	+2	5-10 cm	25-40 μm
Beta particles	9.11×10^{-27}	±1	0-18 m	0-1 cm
Gamma rays and x rays	---	0	0.1-100 m+[b]	1 mm-1 m+[b]
Neutrons	1.67×10^{-24}	0	0-100 m	0-100 cm

(a) The unit charge is approximately 1.6×10^{-19} coulombs.
(b) There is no real endpoint for electromagnetic radiation; however, its intensity is reduced as it travels farther and passes through materials.

(1) Alpha Particles. Alpha particles are emitted only from very heavy nuclei that have an atomic number, Z, of 82 or more, except in some artificially produced nuclides. An alpha particle (α) is a helium nucleus. It has two protons and two neutrons and a net charge of +2. When a parent nucleus decays by alpha emission, the atomic number of the daughter nucleus is two less than that of the parent, and the mass number, A, of the daughter nucleus is four less than that of the parent. This reaction is summarized in Table 1.2.

(2) Beta Particles. Beta particles result when a proton is converted to a neutron or a neutron is converted to a proton in the nucleus. These transitions help an unstable nucleus establish a more favorable neutron-proton ratio. After such a transition, two types of particles are ejected from the nucleus: a neutrino and a beta particle. A neutrino, symbolized υ, has no charge and essentially no mass and travels at the velocity of light. The neutrino does not easily interact with matter and presents no radiation hazard. A beta particle can have either positive or negative charge, depending on the type of transition in the nucleus. If the beta particle is negatively charged, it is an electron; if positively charged, it is a positron.

When the nucleus has an excess number of neutrons, it undergoes a neutron-to-proton transition, and a negatively charged beta particle, or electron, is

TABLE 1.2. Effect of Common Decay Types on the Parent Nucleus

Decay Type	Change from Parent to Daughter Nucleus: Atomic Number, Z	Neutron Number, N	Mass Number, A	Reaction Summary[a]
Alpha	-2	-2	-4	$^{A}_{Z}X \rightarrow {}^{4}_{2}\alpha + {}^{A-4}_{Z-2}X + h\upsilon$
Beta negative (electron)	+1	-1	No change	$^{A}_{Z}X \rightarrow \beta^{-} + {}^{A}_{Z+1}X + h\upsilon$
Beta positive (positron)	-1	+1	No change	$^{A}_{Z}X \rightarrow \beta^{+} + {}^{A}_{Z-1}X + h\upsilon$
Electron capture	-1	+1	No change	$^{A}_{Z}X + e^{-} \rightarrow {}^{A}_{Z-1}X + h\upsilon$

(a) $h\upsilon$ = energy of protons (see Equation (1.5)).

ejected. As shown in Table 1.2, this beta negative decay results in the atomic number, Z, of the nucleus increasing by one. The mass number, A, remains constant.

Proton-to-neutron transition occurs when the nucleus has an excess number of protons. In this case, a positively charged beta particle, or positron, is ejected in what is called beta positive decay or positron decay. As a result of this decay, the atomic number Z decreases by one while the mass number A remains constant.

Sometimes a nucleus has an excess number of protons but is unable to emit a positron. In this case, the nucleus captures an orbiting electron, which combines with a proton to form a neutron. This process is called electron capture decay, and the resulting nuclear change is identical to that of positron emission: the atomic number decreases by one and the mass number remains constant. Because an electron has been removed from its orbit, x rays are produced as the electrons become rearranged (see Section (4) below).

Beta particles are emitted from the nucleus with a spectrum of energies. The beta particle and the neutrino are emitted together and share a given

amount of energy, but the sharing is not in a constant ratio. The beta particle may therefore be ejected from the nucleus with essentially no kinetic energy, or with a high kinetic energy. The average energy of the beta particles emitted is about one-third of the highest kinetic energy for beta particles. Tables of beta energies indicate the highest energy level for betas, but only a small fraction of beta particles possess this highest energy level.

(3) Gamma Rays. When radioactive decay results in the emission of a particle from the nucleus, the nucleus is often left in an excited state. The excited nucleus then releases its excess energy in the form of gamma rays (photons, or wave packets of electromagnetic radiation) until the ground energy state of the nucleus has been reached. Sometimes the energy is emitted in one jump; at other times it is emitted in a series of jumps. The number and energy of gamma rays given off following particle emission is characteristic of a given radionuclide.

Gamma rays are usually emitted immediately after the particle is ejected, but sometimes the nucleus remains in a high-energy state for a measurable period of time, up to several hours. The excited nucleus is then in an unstable, transient condition and is called an isomer of the ground-state nucleus. Isomers are nuclei that are identical to each other in all respects except for their energy state. The excited state is designated by writing "m" after the mass number of the nuclide; for example, ^{99m}Tc is an isomer of technetium-99 and decays to ^{99}Tc by releasing a gamma ray.

(4) X Rays. The capture of an orbital electron by a nucleus with excess protons (electron capture decay) results in a vacancy in the shell that the electron occupied. The shell most commonly vacated is the K shell, that closest to the nucleus. Because an electron from an outer shell jumps down to fill the vacancy, electron capture is always accompanied by the emission of characteristic radiation in the form of x rays. Like gamma rays, x rays are photons, or quanta of electromagnetic radiation.

(5) Neutrons. Neutrons are not emitted from the more common radionuclides. Some of the heavier radionuclides emit neutrons by spontaneous fission, or splitting of the nucleus. The most common example of a spontaneously fissioning radionuclide is californium-252 (^{252}Cf). Other sources of neutrons are listed below.

1. Some isotopes of boron, beryllium, lithium, sodium, fluorine, and other elements with a low atomic number emit neutrons when irradiated by alpha particles or gamma rays. These neutron sources are prepared by mixing a radioactive nuclide and a finely divided powder of the target substance. Examples of neutron sources are the mixed powders ^{241}Am:Be (americium and beryllium) and ^{210}Po:Be (polonium and beryllium), and the chemical compound $^{239}PuF_4$ (plutonium fluoride). Neutron sources are kept in sealed metal containers, and the neutrons emitted have a spectrum of energies.
2. When high-speed charged particles irradiate a suitable target material, the resulting nuclear reactions yield neutrons. These high-speed particles, or accelerator sources, can be used to produce neutrons of nearly the same energy.
3. The fission process in nuclear reactors produces large numbers of neutrons with a spectrum of energies.

1.2.2 Decay Pathways

A radionuclide can undergo radioactive decay via more than one decay pathway. Each decay pathway consists of the emission of a particle followed, in most cases, by the emission of one or more gamma rays. Pathways differ in the manner in which the energy of decay is distributed among the particle emitted and the subsequent gamma rays. For example, radium-226 (^{226}Ra) can decay by five pathways. The most common pathway is the emission of an alpha particle with 4.78 MeV of kinetic energy; the resulting (daughter) nucleus, radon-222 (^{222}Rn), is not in an excited state, so no gamma is emitted. The next most common pathway is the emission of an alpha particle with a kinetic energy of 4.60 MeV. The ^{222}Rn daughter nucleus is in an excited state, and a gamma ray is emitted. For three additional pathways with alpha energies of

4.34, 4.19, and 4.16 MeV, the emission of gamma rays follows. A single nucleus can decay by only one of the various pathways, but because there are five potential pathways, it is sometimes said that ^{226}Ra has five alphas, or five potential alpha energies. Appendix A contains a more detailed discussion of decay pathways.

1.2.3 Quantification of Radioactivity

Radioactive materials are not always measured by their mass or the number of atoms present. They are usually measured by the number of nuclear decays or disintegrations occurring in a sample at any time. The number of disintegrations occurring in a sample per unit time is the activity of the sample. The traditional unit of activity is the curie, abbreviated Ci. One curie is the amount of material undergoing 3.7×10^{10} disintegrations per second (dps). Several fractions of the curie are in common usage: the microcurie, abbreviated μCi, is one millionth of a curie (3.7×10^{4} dps), and the picocurie, abbreviated pCi, is 3.7×10^{-2} dps or 2.22 disintegrations per minute (dpm). The international system (SI) unit of activity, the becquerel, abbreviated Bq, is 1 dps.

A radionuclide's activity, A, is related to its decay constant, λ, and the number of radioactive atoms present, N, by the equation $A = \lambda N$. Remember that $\lambda = (\ln 2)/t_{1/2}$. From this equation, we learn that for a given sample activity, fewer radioactive atoms are present if the half-life is short than if the half-life is long.

The activity represents the disintegration rate of the sample; for every disintegration, one or more radiations may be emitted. As a result, two samples of equal activity may emit different amounts of radiation. For example, each disintegration of cobalt-60 (^{60}Co) involves the emission of one electron followed by two gammas, whereas each disintegration of ^{3}H and ^{14}C involves the emission of only one electron, without gammas.

The activity of a radioactive sample is directly related to the number of radioactive atoms present. For this reason, the activity of the sample decreases exponentially as the number of radioactive atoms present decreases. That is, the activity of a sample of a radionuclide can be determined at any time using the following equation:

$$A = A_0 e^{-\lambda t} \qquad (1.6)$$

where A = the activity present at time t

A_0 = the activity originally present

e = the base of the natural logarithms (2.71828)

λ = the decay constant of the radionuclide = $(\ln 2)/t_{1/2}$ = $0.693/t_{1/2}$

t = the elapsed time.

The specific activity is defined as the activity of 1 gram of radioactive material and is usually expressed as Ci/g of the material. The specific activity of a radionuclide is inversely proportional to its half-life; that is, a radionuclide that has a short half-life will have a higher specific activity than a radionuclide that has a long half-life.

Section 1.3 INTERACTIONS OF RADIATION WITH MATTER

All radiation, whether particulate or electromagnetic, possesses energy. The reduction of this energy, or of the radiation's intensity, as it passes through some matter is called attenuation. Attenuation is a combination of two processes, absorption and scattering. Absorption involves the dissipation of the radiation energy into the absorbing medium; scattering involves the deflection of the radiation from its original path. The mechanisms of radiation attenuation are described in this section.

1.3.1 Alpha and Beta Particles

The transfer of energy from radiation to the atoms of an absorbing material can occur by several processes. Alpha and beta particles transfer energy primarily by the absorption processes of excitation and ionization.

A. Energy Transfer Processes. Excitation is the raising of an electron in an atom or molecule of the absorbing material to a higher energy level without the electron being ejected from the atom or molecule. The electron

then returns to its original energy state, at the same time releasing electromagnetic radiation in the form of light or x rays.

Ionization involves the transfer of sufficient energy to an electron to remove it from the electronic structure of the atom or molecule. Depending on the degree of the interaction, the ejected electron may possess anywhere from a negligible up to a very large amount of kinetic energy. If the electron is given sufficient kinetic energy as it is ejected, it may cause excitation and ionization in other atoms of the absorbing material it is passing through; it is then termed a delta ray. The isolated electron and the remaining atom together are called an ion pair. The average number of ion pairs formed by radiation per unit length of the matter it passes through is called the specific ionization of the radiation.

As alpha and beta radiations move through an absorbing medium and their energy of motion is transferred to the orbiting electrons of the absorbing medium by excitation and ionization, the alpha and beta particles gradually lose all kinetic energy until virtually no energy is left. The rate of energy loss as the radiation traverses a material is called the linear energy transfer (LET) of the radiation and is measured in joules per meter (J/m). (Historically, LET has also been expressed in terms of keV/μm.) In general, the higher the LET of the radiation, the shorter its range (the distance it travels) and the greater the biological hazard it presents because all its energy is deposited over a smaller volume of tissue.

B. Alpha Particle Interactions. An alpha particle is emitted from the nucleus of a radioactive atom with a velocity about one-twentieth that of light. Because of its low velocity and double positive charge, the alpha particle interacts readily with atomic electrons by excitation and ionization, and has a very high specific ionization and LET. The alpha particle loses kinetic energy very rapidly, so it has a low penetrating ability and travels only a few centimeters in air. (Refer back to Table 1.1.) An alpha particle can usually be stopped by several sheets of paper or a sheet of aluminum foil. After an alpha particle loses all of its energy, it attracts two electrons and becomes a helium atom.

The range of an alpha particle in tissue is 35 to 70 μm, depending on its original energy. Because this range is about the same as the thickness of the dead skin layers on the human body, an alpha-emitting radionuclide is considered to present little hazard outside of the body. There are a few exceptions to this general rule. First, if the skin is broken, living tissue may be irradiated. Second, in the case of the eye, the living tissues are very close to the surface and can be harmed by alpha radiation.

The greatest biological hazard due to alpha-emitting radionuclides occurs when the material enters the body by inhalation or ingestion. In this case, there are no dead cells to absorb the energy, and living tissue is irradiated.

C. Beta Particle Interactions. Beta particles are emitted from the nucleus with a velocity much greater than that of alpha particles, even approaching the velocity of light. Beta particles are more penetrating than alpha particles and can travel up to 18 meters in air, depending on their energy; however, they can be stopped by a few millimeters of materials such as plastic, aluminum, and iron.

Beta particles lose their energy primarily by interacting with the electrons of the absorbing medium. Beta particles can also slow down in the electrical field of atomic nuclei to produce x rays. The x rays produced in these interactions are called bremsstrahlung (from the German word for "braking," so named because this radiation results from the slowing down of beta particles). The energy of the bremsstrahlung may range from negligible up to the energy of the beta particle. The probability of this interaction occurring is greater for radionuclides that emit high-energy beta particles, such as phosphorus-32 (^{32}P) and yttrium-90 (^{90}Y), and for absorbing materials with a high atomic number, such as iron or lead, than for radionuclides that emit beta particles with lower energies and for absorbing materials with a low atomic number. The radiation produced is identical in all respects to gamma or x radiation of the same energy. Bremsstrahlung photons can present a significant radiation hazard when radionuclides that emit high-energy beta particles are stored in metallic containers. In order to reduce the production of bremsstrahlung, emitters of high-energy beta particles should be kept in thick-walled plastic containers. The plastic containers may then be placed in iron or lead

containers to protect against any photons other than bremsstrahlung that may be emitted. Bremsstrahlung is not produced in any significant amount in biological materials because the elements of which human tissues are composed have low atomic numbers.

The LET of beta particles is much lower than that of alpha particles because betas have only a single charge and travel at high velocities. In many cases, beta radiation is considered to be only a slight hazard outside the body, because even though betas with an energy higher than 70 keV can penetrate to living skin tissue, they still cannot reach the major organs of the body. However, beta particles can cause severe damage to the skin and the eye. Thus we can say that beta particles present more of an external hazard than do alpha particles.

Inside the body, beta radiation is less hazardous than alpha radiation. Because the LET of beta particles is less than that of alpha particles, the energy deposited by the beta radiation is dissipated over a larger volume than is the energy deposited by alpha radiation.

After a negatively charged beta particle (an electron) loses all of its kinetic energy, it becomes attached to a positive ion, becoming an orbital electron. A postively charged beta particle (or positron), on the other hand, is antimatter and cannot exist for long in nature. After it loses all of its kinetic energy, it fuses (coalesces) with an electron, the two particles annihilate each other, and their mass is converted into energy. This energy is observed as two photons, called annihilation radiation, each of which has 0.511 MeV of energy. The two photons are emitted in opposite directions.

1.3.2 Photons

Gamma rays and x rays are both forms of electromagnetic radiation and they have identical properties. The only difference between them is that gamma rays are emitted from the nucleus and x rays arise from processes outside the nucleus. X rays produced as a result of radioactive decay tend to have lower energies than gamma rays, while x rays produced by x-ray machines can have energies much higher than the energies of gamma rays.

A. Energy Transfer Processes. Ionizing photons interact with matter by three major mechanisms: the photoelectric effect, the Compton effect, and pair production. Which interaction takes place depends on the photon energy and on the atomic number, Z, of the absorbing medium. Figure 1.6 shows the relative importance of these interactions as a function of Z and photon energy. The end result of all three interactions is the production of high-energy electrons, which interact with matter in the same way beta particles do.

The photoelectric effect is an interaction between a photon and an orbital electron. In this process, the photon ceases to exist and its energy is transferred to the electron, which is ejected from the atom with a kinetic energy equal to the energy of the photon minus the binding energy of the electron. The photoelectric interaction is dependent on the energy of the photon and strongly dependent on the atomic number of the absorbing material. It is most likely to occur in high-Z materials, such as iron and lead, and at low photon energies, less than 100 keV (0.1 MeV). The photoelectric effect is not an important interaction in biological systems, which are made up primarily

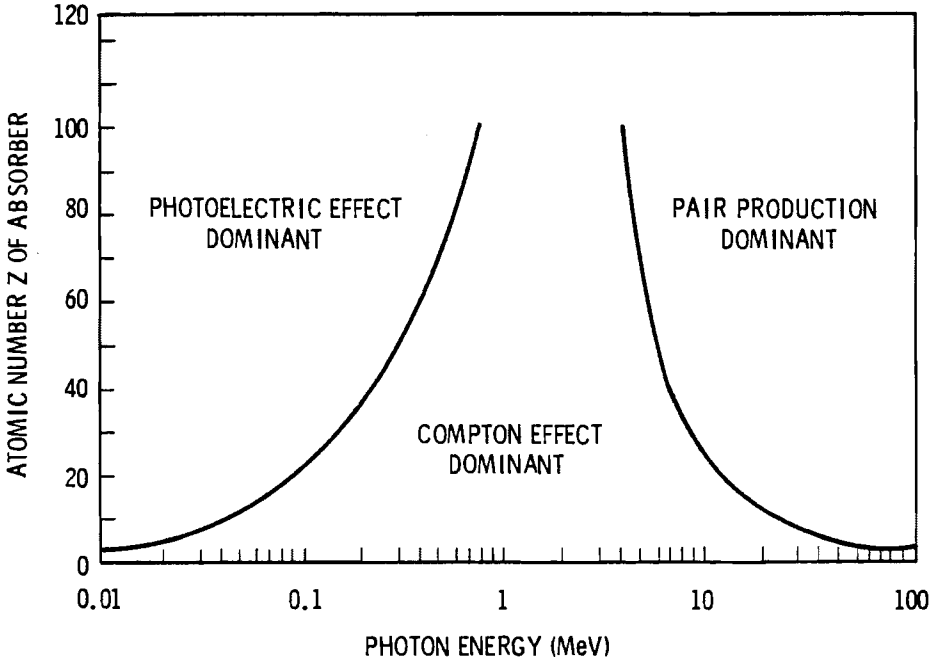


FIGURE 1.6. Relative Importance of the Photoelectric Effect, the Compton Effect, and Pair Production

of carbon, oxygen, hydrogen, and nitrogen, all low-Z elements. However, it is important in high-Z materials and is useful for identifying and quantifying gamma-emitting radionuclides.

The Compton effect (or Compton scattering) is the predominant interaction between biological materials and photons from 30 keV (0.03 MeV) to 10 MeV. In the Compton interaction, the photon interacts with an orbital electron that is not tightly bound to the nucleus. The photon transfers part of its energy to the electron, which is ejected from the atom. The photon is then scattered by (deflected from) the atom at reduced energy. The scattered photon can go on to interact with electrons of other atoms.

High-energy photons can interact with the electrical field of the atom's nucleus via pair production. In this process, when a photon passes close to the nucleus of an atom, the photon ceases to exist, and 1.02 MeV of energy is converted into an electron (negatron) and a positron. If the original photon had an energy greater than 1.02 MeV, the remaining energy is shared by the electron and the positron in the form of kinetic energy. This interaction, which does not occur if the original photon energy is less than 1.02 MeV, is of greatest importance in high-Z materials and does not often occur in biological tissue.

B. Photon Interactions. Photon-emitting radionuclides outside the body can present a severe hazard for several reasons. First, photons can penetrate through thick layers of lead and concrete, so it is difficult to shield the body against them. Second, they can penetrate great distances through air and may therefore constitute a hazard even far from a source of radiation. Finally, photons can easily penetrate the skin and can irradiate organs within the body; in fact, they can irradiate the whole body. However, photons are less of an internal hazard than either alpha or beta radiation because they have a low LET and distribute their energy throughout the body rather than concentrating it in one small area.

1.3.3 Neutrons

Neutrons, like photons, are very penetrating. Because they have no electrical charge, neutrons, unlike other types of radiation, do not interact with electrons. They do interact with atomic nuclei, yielding charged particles

that can then deposit energy in an absorbing medium by excitation and ionization. Neutrons are not stable outside the nucleus. They have a half-life of 10.6 min and decay to a proton and an electron.

Neutrons can be classified by their energies; one such classification scheme is shown in Table 1.3. All neutrons are fast neutrons when produced. Neutrons that have lost most of their energy are called thermal neutrons. Thermal neutrons receive their name from the fact that they are in approximate thermal equilibrium with their environment.

A. Energy Transfer Processes. To a large extent, the type of interaction that a neutron undergoes depends on its energy. Most fast neutrons lose their energy by colliding with nuclei in what are termed elastic collisions or "billiard ball" collisions. For neutrons with energies between 100 keV and 20 MeV, this is the predominant interaction with biological materials. When incident neutrons collide with the nucleus of an atom of the absorbing material, part of the neutron's kinetic energy is transferred to the nucleus and part is retained by the deflected neutron, which may then undergo additional collisions until it has lost virtually all of its energy.

Fast neutrons may also lose their energy by inelastic scatter. In this process, a neutron transfers part of its kinetic energy to the nucleus of an atom. The nucleus is then in an excited state and emits a gamma ray to return to its ground state. Inelastic scatter is a phenomenon more closely associated with high-Z absorbers, such as iron or lead, than with low-Z absorbers, such as hydrogen or carbon.

TABLE 1.3. Classification of Neutrons

Neutron Classification	Energy
Thermal neutrons	$\leq$0.025 eV
Slow neutrons	0.025 eV to 100 eV
Intermediate neutrons	100 eV to 10 keV
Fast neutrons	$\geq$10 keV

A neutron may enter the nucleus of an atom and undergo radiative capture. The resultant nucleus, one mass unit heavier than the original, is in an excited state and emits a gamma ray. Because the gamma rays arising from this type of interaction may have energies up to several MeV, they contribute to the shielding difficulties encountered with neutrons.

In radiative capture with particle emission, a neutron may be captured by a nucleus that subsequently ejects a charged particle, for example, a proton or an alpha particle. This interaction is used to infer the presence of neutrons and to produce radioactive isotopes.

The capture of a neutron by certain heavy nuclei may result in fission, the splitting of the nucleus into two lighter nuclei of approximately equal mass, called fission fragments. As the nucleus disintegrates, an average of two or three neutrons is emitted. If one of these causes a subsequent fission, a steady-state chain reaction may take place. Some nuclei undergo fission after absorbing a thermal neutron, others after absorbing a fast neutron. Fission fragments are radioactive and present a radiation hazard of their own.

B. Neutron Interactions. In soft tissues of the body, the predominant interaction is collisions between incident neutrons and hydrogen nuclei, which are single protons. This interaction is important because a large fraction of the neutron energy is transferred to the proton, since its mass is almost the same as that of the neutron. Furthermore, hydrogen is the most abundant atom in the tissues. The protons that are set into motion by this process lose energy by the excitation and ionization of atoms as they pass through biological material. These protons have a high LET and can produce significant biological damage.

At kinetic energies below a few hundred keV, radiative capture of neutrons becomes important. The capturing nuclei are primarily those of hydrogen and nitrogen. Neutron capture by hydrogen, ^{1}H, results in the emission of a 2.2-MeV gamma ray; at the same time, the ^{1}H nucleus is converted to a ^{2}H nucleus. Neutron capture by a ^{14}N nucleus leads to the emission of a 660-keV proton and the transformation of the ^{14}N nucleus to a ^{14}C nucleus. The probability of neutron capture by other elements in the body is small.

If the body is exposed to a high concentration of neutrons, two reactions occur that can be used to estimate the neutron exposure: sodium in the tissues and blood is converted from ^{23}Na to ^{24}Na, and sulfur in the hair changes from ^{32}S to ^{32}P. The radiation dose from these radioactive nuclides is small compared to the radiation dose received from the large number of neutrons required to activate an appreciable number of target atoms.

Section 1.4 RADIATION QUANTITIES AND UNITS

Radiation measurements and units of radiation dose are based primarily on the energy deposited by radiation as it travels through matter. The International Commission on Radiation Units and Measurements (ICRU) selects and defines the units and quantities of radiation. Information provided in this section is based on ICRU Report 33, Radiation Quantities and Units (ICRU 1980).

1.4.1 Exposure

The term "exposure" has two levels of meaning. The first level is that of an object or person being subjected to the action of radiation. It is in this context that the word is most commonly employed, especially by the public. For example, a person might say "I was exposed to neutrons." In radiation protection, on the other hand, the term exposure is used to quantify the amount of x or gamma radiation present. In a given situation, the meaning of the word is determined from the context in which it is used.

In the context of radiation protection, exposure is a measure of the ionization produced by x or gamma radiation in air. The ionization is measured by collecting all the electrons liberated by the photons through photoelectric, Compton, and pair production interactions. Note that exposure, in this sense of the word, is defined only for x and gamma radiation, and that the measurement must be made in air. In practice, exposure is difficult to measure precisely when the photon energies involved are below a few keV or above a few MeV.

The special unit for exposure is the roentgen (R). One roentgen is equal to 2.58×10^{-4} coulomb/kg of air. This seemingly arbitrary value is equivalent to 1 electrostatic unit of electricity (esu) per cubic centimeter of air at standard temperature and pressure (STP), which was the original definition of the roentgen. One roentgen results in the production of 2.08×10^{9} ion pairs/cc of dry air at STP. The energy required to produce these ion pairs is approximately 87.7 ergs/g of air.

1.4.2 Absorbed Dose

The absorbed dose describes the quantity of radiation energy transferred to any absorbing material (tissue, air, shielding, etc.). The ICRU has defined absorbed dose, symbolized D, as

$$D = \frac{d\bar{\varepsilon}}{dm} \tag{1.7}$$

where $d\bar{\varepsilon}$ is the mean energy imparted by ionizing radiation to matter of mass dm. The advantage of absorbed dose as a measure, as compared with exposure, is that absorbed dose can be applied to any radiation and any absorbing medium. The unit of absorbed dose is called the rad and is equal to 100 ergs/g of the absorbing material. In the international system of units (the SI unit), the absorbed dose is the gray (Gy) and is equal to 1 joule/kg.

$$1 \text{ rad} = 100 \text{ ergs/g} = 10^{-2} \text{ J/kg} = 0.01 \text{ Gy}$$

For x and gamma rays, the exposure (expressed in units of roentgens) can be related to the absorbed dose in tissue (expressed in rad) by the equation

$$D_{tissue} \cong 0.97 \, X \tag{1.8}$$

where X is the exposure in roentgens. This equation holds for x or gamma radiation of energies from 0.1 to 10 MeV. From this equation, the absorbed dose, in rad, to an individual exposed to x or gamma radiation can be determined by measuring the exposure, in roentgens, at the location where the

individual was exposed. Potential radiation doses to individuals working with sources of x or gamma radiation can also be estimated, as will be discussed in subsequent chapters.

1.4.3 Relative Biological Effectiveness

Because radiations interact with matter in varying ways, equal doses of different types of radiation do not always produce equal biological effects. When comparing the effects of different radiations, it is customary to use 250-kV x rays as the standard. This radiation was chosen as a standard because its effects were well documented and it was the only type of radiation widely available at the time this convention was adapted.

The formal definition of relative biological effectiveness (RBE) is as follows: the RBE of a test radiation is defined by the ratio D_{250}/D_r, where D_{250} is the absorbed dose of 250-kV x rays and D_r is the absorbed dose of the test radiation required to produce an equal biological effect.

The RBE is often used in radiation biology, but the concept has limited usefulness in radiation protection because the RBE of a given radiation is influenced by the specific conditions of the experiment. The dose rate used, the dose fractionation (or the number of increments in which the dose is received), the biological tissue irradiated, and the radiation effect measured all affect the RBE.

1.4.4 Dose Equivalent

The results of biological experiments have shown that the absorbed dose by itself is insufficient for predicting either the probability of deleterious health effects from irradiation under unspecified conditions, or the severity of such effects. The RBE of radiation is also not useful, primarily because of the many factors that can influence it. Consequently, an additional quantity has been defined.

This quantity, a quality factor, Q, accounts for the different biological effects that result from the ways various types of radiation distribute energy. The values of Q are defined as a function of the radiation's LET in water and are based on relevant values of RBE. Table 1.4 shows the recommended values

TABLE 1.4. Relationship of LET and Quality Factor

LET (keV/μm)	Q
≤3.5	1
7	2
23	5
53	10
175	20

of Q as a function of LET in water. It is possible to find exact quality factors based on the LET by interpolating the values given in the table. However, it is common practice to use the recommended values for different types of radiation, as given in Table 1.5.

The absorbed dose and the quality factor are incorporated into a third quantity, called the dose equivalent. The dose equivalent, H, at a point in tissue is given by the equation

TABLE 1.5. Recommended Values of Q for Different Types of Radiation(a)

Radiation	Q
X rays, gamma rays, and electrons	1
Neutrons, protons, and singly charged particles with a rest mass greater than 1 atomic mass unit and with an unknown energy	10
Alpha particles and multiply-charged particles (and particles of unknown charge) with an unknown energy	20

(a) Based on Report No. 39 of the National Council on Radiation Protection and Measurements (NCRP 1971).

$$H = DQN \qquad (1.9)$$

where H = dose equivalent (rem)
D = absorbed dose (rad)
Q = quality factor
N = modifying factors.

The numerical value of N is generally considered equal to 1. The special name for the unit of dose equivalent is the rem. The SI unit for dose equivalent is the sievert (Sv), which equals 1 J/kg. If the absorbed dose is given in units of gray, then the dose equivalent is in units of sievert.

The dose equivalent is a valuable term because the varying biological effects of different types of radiation are accounted for through the quality factor, Q. Therefore, the effect of 1 rem (or 0.01 Sv) of radiation is nearly the same for all types of radiation. This equivalence permits the addition of dose equivalents when several radiations are involved.

Section 1.5 BIOLOGICAL EFFECTS OF RADIATION

Just as atoms are the basic building blocks of elements, cells are the basic unit of the human body. The body is composed of millions of cells, each with a specific job to do to keep us alive and well. When radiation transfers energy to cells, primarily by the processes of excitation and ionization, it can disturb the cells so they can no longer perform their original functions.

The cells that make up the various tissues of the body do not have identical functions or appearances. For example, the cells that make up nerve tissue look and act differently from those that make up muscle tissue. Each type of cell may react differently to radiation. Some cells are more radiosensitive than others (that is, susceptible to radiation injury). In the body, the most radiosensitive cells are the blood-producing and the reproductive cells. Muscle, nerve, and bone cells are the least radiosensitive. Radiation has two main types of effects on biological systems: genetic effects and somatic effects.

1.5.1 Genetic Effects

Genetic effects are biological effects of radiation that result in mutations, or changes, in the genes of reproductive cells and that are expressed in the descendants of the exposed individual. Mutations occur in all living organisms. They can be induced by agents such as radiation or chemicals, or they can occur spontaneously, without any outside alteration in the physical or chemical environment. Genetic effects of radiation appear as birth defects in the offspring of the irradiated individual and in succeeding generations, as demonstrated in experiments involving thousands of irradiated animals. These effects have not been observed in human populations, perhaps because few people have received the high doses thought to cause such effects.

1.5.2 Somatic Effects

Somatic effects are biological effects of radiation that are expressed in the exposed individual. The somatic effects of radiation can be divided into prompt effects and delayed effects.

A. Prompt Effects. Prompt effects are observed shortly after an individual receives an acute radiation dose, a very large dose received in a very short time period. Prompt effects are generally associated with a threshold; that is, if the radiation dose is below a certain level, no effect is noticed, but if the dose exceeds that level, most people suffer an effect. Prompt effects tend to be short-term. The short-term effects of acute exposure to high levels of ionizing radiation are well known from observations of individuals exposed during atomic warfare, medical treatments, or industrial accidents. These effects may include nausea, fatigue, blood disorders, intestinal problems, temporary loss of hair, skin burns, and in extreme cases, death. Table 1.6 shows the effect of an acute whole-body exposure in relation to dose, and Table 1.7 shows the effect of partial-body irradiation in selected organs. Note that whole-body irradiations are much more dangerous than partial-body irradiations. If radiation safety standards are met, there is no reason for any individual to experience prompt radiation effects.

B. Delayed Effects. Delayed effects can result from an acute radiation dose and are the major effects of a chronic radiation dose. A chronic radiation dose is the continuous or repeated subjection of an individual to

TABLE 1.6. Dose-Effect Relationship for Acute Whole-Body Irradiation

Acute Dose (rem)	Nature of Effect
5-23	Minimal dose detectable by chromosome analysis or other specialized analysis
25-125	Slight blood changes
75-125	Minimal acute dose likely to produce vomiting in about 10% of people so exposed
150-200	Temporary disability, blood changes
300-500	Mean lethal dose

TABLE 1.7. Dose-Effect Relationship for Acute Partial-Body Irradiation

Acute Dose (rad)	Organ	Effect in Relevant Organs
50	Testis	Temporary sterility
200	Ovary	Temporary amenorrhea, sterility
500	Skin	Temporary reddening and loss of hair
800	Testis	Permanent sterility
	Ovary	Permanent menopause, sterility
2000	Liver	Hepatitis
2500	Skin	Temporary ulceration and permanent loss of hair

radiation at low dose rates over a long period of time. The primary delayed somatic effects are the development of cancer and, to a lesser extent, the production of cataracts. As opposed to prompt effects of radiation, delayed effects are associated not with thresholds but with probabilities of occurrence: as the radiation dose increases, the likelihood of observing an effect increases. A relationship between radiation dose and cancer induction has been shown from studies of 1) Japanese survivors of the atom bomb; 2) the Marshall

Islanders, who were exposed to fallout from weapons tests; 3) uranium miners; and 4) radiation therapy patients who received excessive doses in the early part of the century. These situations all involved much higher radiation doses than those today's radiation workers can legally receive.

C. Relationship Between Exposure and Delayed Effects. The exact relationship between chronic low-level exposure and delayed effects is difficult to establish for two reasons. First, effects such as cancer can be caused not only by radiation but also by other agents in the environment, such as cigarette smoke or chemical pollutants. Second, long periods of time may elapse between an exposure to radiation and the observation of any effects.

We do not yet know how radiation causes cancer. However, most diseases are caused by the interaction of a variety of factors, including general physical condition, inherited traits, age, sex, and exposure to outside agents. It is impossible to know whether a given cancer is caused by radiation or some other agent. However, we do know that an increased incidence of cancer is observed in groups of highly exposed people. Although several studies have been performed, there is no firm evidence that exposure to radiation at currently accepted levels results in an increased incidence of cancer.

1.5.3 Environmental Dose and Occupational Dose Limits

Individuals who work with radiation receive a radiation dose from the environment as well as from their workplace. Table 1.8 shows the estimated average individual dose in millirem from natural background radiation and other sources. The table indicates that the average individual in the United States receives a dose of about 200 mrem of radiation each year from sources that are part of our natural and man-made environment.

The standards of radiation dose suitable to the workplace are set by federal regulations. Table 1.9 lists the dose standards for various parts of the body. These standards do not represent boundaries between safe conditions and harmful or lethal conditions. Rather, they represent dose levels for which regulators consider there is sufficiently small probability of radiation effects. Because the likelihood of causing an effect increases gradually with increasing dose, it is wise to keep the actual radiation dose as low as is reasonably achievable (ALARA).

TABLE 1.8. U.S. General-Population Dose Estimates (1978)[a]

Source	Average Individual Dose (mrem/yr)
Natural background	100
Medical	90
Release of radioactive material by mining, milling, etc.	5
Nuclear weapons development (primarily fallout)	5 to 8
Nuclear energy	0.28
Consumer products	0.03
TOTAL	200 mrem/yr

(a) Interagency Task Force on the Health Effects of Ionizing Radiation, 1979.

TABLE 1.9. Maximum Dose Equivalent Per Calendar Quarter[a]

Organ	Amount (rem)
Whole body; head and trunk; active blood-forming organs; lens of eyes; gonads	1.25
Hands and forearms; feet and ankles	18.75
Skin of whole body	7.50

(a) AR 40-14.

Section 1.6. PROPERTIES OF RADIOACTIVE MATERIALS IMPORTANT IN THE DEVELOPMENT OF RADIATION PROTECTION PROCEDURES

Several properties of radioactive materials play a key role in the development of radiation protection procedures. These include the distinction between external and internal exposure, and the properties of dispersibility, chemical toxicity, radiotoxicity, and criticality.

1.6.1 External Versus Internal Exposure

The extent to which radiation causes biological effects depends in part on whether the body is exposed externally or internally, and on the types of radiation involved in the exposure.

A. External Exposure. External exposure results from exposure to a source of ionizing radiation outside the body. Sources of external exposure can be divided into two classes: penetrating and nonpenetrating radiations. Penetrating radiations--gamma rays, x rays, and neutrons--have sufficient energy to pass through the surface of the skin and interact with internal body tissues. Nonpenetrating radiations--alpha particles and low-energy beta particles--interact only with the skin surface. Therefore, from the standpoint of external exposure, penetrating radiation is a greater hazard than nonpenetrating radiation.

The principles and procedures that minimize external exposure, and the calculation of external dose, are discussed in Chapter 6.

B. Internal Exposure. Radioactive materials can be taken into the body by ingestion, inhalation, and absorption through pores of the skin or through breaks in the skin. Once in the body, these materials may be deposited in various organs and constitute a source of internal exposure.

A stable isotope and a radioactive isotope of the same element have identical chemical behavior in the body. The chemical characteristics of an isotope or nuclide determine the organ in which it is deposited as well as the rate at which it is excreted from the body. If a radionuclide has no stable counterpart in the body, it follows the metabolism and excretion pattern of another element with similar chemical properties. For example, strontium is

not normally found in large quantities within the body. However, the chemical properties of strontium are similar to those of calcium. Thus, strontium that enters the body behaves much as calcium does and may be deposited in the bone.

The radiological hazard associated with internal exposure depends upon the type of radiation emitted by a radionuclide, the radiosensitivity of the organ in which it is deposited, and the physical properties of the radionuclide (e.g., its solubility and particle size). Of the various types of radiation, alpha particles are usually considered the greatest internal hazard.

The calculation of internal dose is discussed in Chapter 5 along with two principles that are important in making those calculations, the principles of maximum permissible concentration (MPC) and the critical organ. Procedures for minimizing internal exposure to radiation are also discussed in Chapter 5.

1.6.2 Dispersibility

The physical form of a radioactive material and its intended use influence how much it will scatter, or disperse. For example, a radioactive powder has a greater chance of being scattered over a wide area than does a sealed source. Conditions of use under which various forms of radioactive material are nondispersible, of limited dispersibility, dispersible, or highly dispersible are listed below. Engineered safeguards and administrative controls for each of these types of materials are discussed throughout the manual.

A. Nondispersible

1. nondestructive use of encapsulated or sealed sources
2. storage of nonflammable, nonexplosive radioactive materials in sealed containers especially designed for such storage.

B. Limited Dispersibility[(a)]

1. simple operations that can result only in fractional releases of material from a radiation area during credible accidents

---

(a) Criteria used to classify radionuclides in this category are subjective and thus depend in part upon experience and judgment.

2. use of radioactive materials that are strongly bound in a solid matrix or biological system.

C. Dispersible

1. use of unsealed, noncombustible, nonexplosive liquids or compact solids in standard chemical processes or operations.

D. Highly or Readily Dispersible

1. use of radionuclides in hazardous or complex chemical operations
2. use of radioactive powders, gases, vapors, or other aerosols
3. use of radioactive materials in combustible or explosive procedures
4. dry, dusty operations
5. high-temperature or high-pressure operations that may increase the probability of producing radioactive aerosols
6. use of radioactive materials that can ignite spontaneously.

1.6.3 Chemical Toxicity

Chemical toxicity refers to the harm that can be caused by an element because of its chemical nature. Many elements are toxic and can cause severe illness if ingested. Examples of toxic elements include arsenic, which damages blood vessels; cadmium, which is a kidney poison; mercury, which in large doses is a kidney poison and in chronic situations affects the nervous system; and lead, which also affects the nervous system. A radionuclide may be hazardous both because of its chemical nature and because of the radiation it emits. Uranium, for example, is a kidney poison and is also radioactive (it has no stable isotopes). In the case of long-lived isotopes of uranium, it is the chemical rather than the radiation hazard that limits the amount that may safely be ingested. Other radioactive materials, such as plutonium, have negligible chemical toxicity but are considered hazardous because of the amount of radiation damage they can produce. These materials are called radiotoxins.

1.6.4 Radiotoxicity

The term radiotoxicity indicates the relative radiological hazard associated with internally deposited radionuclides. Nuclides that are highly radiotoxic, such as those that emit alpha particles or high-energy beta particles, present the greatest relative health hazard when deposited internally. The level of radiotoxicity strongly dictates the degree of control required in work with radioactive materials. A listing of the relative radiotoxicity of some radionuclides is given in Table 1.10. Note that Group I radionuclides are the least radiotoxic and Group 8 the most radiotoxic.

1.6.5 Criticality

Fission occurs when a heavy nucleus (with an atomic number, Z, of 90 or more) absorbs a neutron and splits into two lighter nuclei, each with about half the mass of the original nucleus. Each fission can also result in the emission of up to eight neutrons, with two and one-half neutrons being the average. A nuclide that is capable of undergoing fission is called a fissionable nuclide. Examples of fissionable nuclides are ^{235}U and ^{238}U. Some nuclides, such as ^{238}U, undergo fission only when they absorb a fast neutron. Other nuclides, such as ^{235}U and ^{239}Pu, undergo fission when they absorb a thermal or slow neutron, and are called fissile nuclides. Materials that contain such nuclides are fissile materials. Natural uranium, which is a combination of ^{235}U and ^{238}U, is a fissile material. Some nuclides, such as ^{235}U and ^{239}Pu, also undergo spontaneous fission; that is, they can split without first having been irradiated by neutrons. Table 1.11 lists some of the more common fissionable nuclides.

After a fission, the neutrons that are released have three possible fates. They may 1) completely escape from the fissile material, 2) be absorbed by nonfissile atoms, or 3) be captured by fissile atoms. If they are captured by fissile atoms, more fissions can occur and more neutrons may be released. The continuing process of fission, release of neutrons, capture of neutrons, and subsequent fission is called a chain reaction.

If the neutrons released by a fissioning atom cause, on the average, less than one subsequent fission, then no chain reaction is possible and the reaction is said to be subcritical. When the neutrons released by each fission cause

TABLE 1.10. Radiotoxicity of Various Nuclides[a]

Radionuclides Grouped by Relative Radiotoxicity	Activity in Curies of Single Inhalation that Results in 15-rem Dose(b) to Critical Organ	Activity in Curies of Single Inhalation that Results in 15-rem Dose(b) to Lung(c)
Group I		
^{3}H	6.15×10^{-2}	---
^{14}C	2.88×10^{-2}	---
Group II		
^{51}Cr	8.84×10^{-2}	5.30×10^{-3}
^{55}Fe	2.17×10^{-3}	2.30×10^{-3}
Group III		
^{35}S	7.23×10^{-4}	6.90×10^{-4}
^{198}Au	7.25×10^{-4}	5.30×10^{-4}
^{47}Ca	2.59×10^{-4}	4.60×10^{-4}
^{132}I	4.50×10^{-4}	---
^{141}Ce	7.06×10^{-4}	4.20×10^{-4}
^{85}Sr	2.00×10^{-3}	2.70×10^{-4}
^{140}La	4.20×10^{-4}	2.60×10^{-4}
^{95}Nb	3.60×10^{-3}	2.30×10^{-4}
^{65}Zn	2.60×10^{-4}	1.50×10^{-4}
^{58}Co	8.40×10^{-3}	1.30×10^{-4}
^{59}Fe	3.00×10^{-4}	1.30×10^{-4}
Group IV		
^{181}Hf	9.94×10^{-5}	1.92×10^{-4}
^{147}Pm	8.90×10^{-5}	2.30×10^{-4}
^{32}P	8.70×10^{-5}	2.10×10^{-4}
^{140}Ba	1.40×10^{-4}	8.60×10^{-5}
^{234}Ba	8.50×10^{-5}	7.30×10^{-5}
^{85}Kr	6.90×10^{-5}	5.80×10^{-5}
^{192}Ir	3.20×10^{-4}	6.90×10^{-5}
^{36}Cl	2.70×10^{-5}	5.30×10^{-5}
^{91}Y	5.00×10^{-5}	7.30×10^{-5}
^{182}Ta	1.10×10^{-4}	5.00×10^{-4}
^{45}Ca	4.30×10^{-5}	2.60×10^{-5}
^{89}Sr	4.00×10^{-5}	8.50×10^{-5}
^{137}Cs	2.60×10^{-9}	2.20×10^{-5}
^{60}Co	2.60×10^{-5}	1.50×10^{-5}
^{144}Ce	1.40×10^{-5}	1.50×10^{-5}
^{126}I	1.40×10^{-5}	7.30×10^{-4}
^{154}Eu	1.30×10^{-5}	1.60×10^{-5}
Group IV (contd)		
^{131}I	1.20×10^{-5}	7.30×10^{-4}
^{170}Tm	3.80×10^{-5}	7.30×10^{-5}
^{82}Br	7.47×10^{-3}	---
Group V		
^{129}I	2.30×10^{-6}	1.60×10^{-4}
^{99}Tc	9.10×10^{-6}	
Group VI		
^{233}Ra	3.90×10^{-6}	5.30×10^{-7}
^{210}Po	1.30×10^{-6}	5.00×10^{-7}
^{227}Th	5.50×10^{-7}	4.60×10^{-7}
^{90}Sr	3.90×10^{-7}	1.30×10^{-5}
^{210}Pb	3.20×10^{-7}	5.30×10^{-7}
^{242}Cm	3.00×10^{-7}	4.60×10^{-7}
^{233}U	7.00×10^{-7}	2.70×10^{-7}
^{235}U (+ 1% ^{234}U)	1.10×10^{-6}	2.60×10^{-7}
^{238}U & U Nat	1.90×10^{-7}	3.00×10^{-7}
^{232}Th & Th Nat	2.25×10^{-9}	2.60×10^{-8}
Group VII		
^{147}Sm	7.70×10^{-8}	6.90×10^{-7}
^{144}Nd	7.70×10^{-8}	7.30×10^{-7}
^{226}Ra	4.90×10^{-8}	1.50×10^{-8}
^{244}Cm	1.10×10^{-8}	2.30×10^{-7}
Group VIII		
^{243}Am	7.60×10^{-9}	2.70×10^{-7}
^{241}Am	6.60×10^{-9}	2.70×10^{-7}
^{237}Np	5.20×10^{-9}	2.70×10^{-7}
^{227}Ac	3.00×10^{-9}	6.90×10^{-8}
^{230}Th	2.80×10^{-9}	2.30×10^{-8}
^{242}Pu	2.50×10^{-9}	8.50×10^{-8}
^{238}Pu	2.20×10^{-9}	8.50×10^{-8}
^{239}Pu	2.00×10^{-9}	8.50×10^{-9}
^{240}Pu	2.00×10^{-9}	8.50×10^{-8}

(a) Brodsky 1965.
(b) Insoluble materials.
(c) 50-year cumulative dose.

TABLE 1.11. Fissionable Materials

Capable of Chain Reaction with Fast and Thermal Neutrons		Capable of Chain Reaction with Fast Neutrons Only
^{233}U	^{243}Cm	^{237}Np
^{235}U	^{245}Cm	^{241}Am
^{239}Pu	^{247}Cm	^{244}Cm
^{241}Pu	^{249}Cf	^{238}Pu
^{242}Am	^{251}Cf	^{240}Pu
		^{242}Pu
		^{238}U

one additional fission, then the reaction is self-perpetuating and the chain reaction is said to be critical. Finally, if the neutrons released from a fissioning atom cause, on the average, more than one subsequent fission, the reaction is said to be supercritical. An unplanned supercritical chain reaction is called a criticality accident.

Criticality accidents are extremely serious because very high levels of gamma and neutron radiation can be produced. That is, lethal doses of radiation can be received in a very short time. For this reason, special efforts are made to reduce the chances of a criticality accident to a very low level. Particularly important is the design of facilities. "Safe-by-geometry" is the best rule to remember in reducing the probability of a criticality accident.

A. The Double-Contingency Rule. One of the most generally accepted approaches to preventing a criticality accident is the double-contingency rule. This rule assumes that a sufficient number of limits and controls exists to ensure that, before a criticality is possible, at least two unlikely, independent, and concurrent changes must occur in one or more of the conditions specified as essential to nuclear safety. This rule calls for controls which ensure that no single mishap can lead to a criticality accident, regardless of the probability that that mishap might occur.

B. Factors That Affect Criticality. Nine physical factors affect the likelihood that an accumulation of fissile material will sustain a chain reaction. Criticality safety programs take account of these factors and employ safeguards based on them to prevent criticality accidents.

(1) Amount. The amount of fissionable material needed to support a chain reaction is the critical mass. If the amount of fissionable material present is small enough, criticality cannot occur no matter what the condition of the other eight physical factors. On the other hand, the greater the amount of fissile material present, the more difficult it is to avoid criticality. Limiting the amount of material present helps ensure a subcritical state. Many safeguards are designed to limit the total amount of fissile material that can be assembled in one place.

(2) Geometry. The size and shape of fissile materials have an important effect on the probability of a chain reaction occurring. Decreasing the distance that neutrons travel within the fissile material decreases the chance that the neutrons will interact to cause a subsequent fission. For this reason, a thin slab of fissile material is unlikely to support fission reactions, but a sphere is most conducive to criticality.

(3) Density. If the density of fissile materials is increased, the fissile atoms are more tightly packed together. This packing reduces the chance that a neutron will escape from the material; thus, the higher the density of the material and the atoms in it, the higher the probability that a fissile atom will capture a neutron in the material and undergo fission.

(4) Moderation. The speed of a neutron affects its chances of being captured by a fissile atom. The faster a neutron travels, the less likely it is to be captured. Thus, the fast neutrons produced by a fission are not likely to cause more fissions until they slow down.

Fast neutrons are slowed down when they collide with, but are not absorbed by, the nuclei of atoms. This slowing-down process is called thermalization, or moderation. Moderation of a neutron increases its chances of being captured and causing a fission. Graphite and hydrogen-containing materials such as paraffin, oil, and water are good moderators. Human tissues are 70% water and thus are good moderators also.

(5) Reflection. Neutrons that escape from fissile materials without causing additional fissions or being absorbed by atoms continue to move away from the materials unless they hit something in their path. Anything placed close to fissile material will tend to bounce (reflect) a few of the neutrons back into the material and give the fissile atoms another chance to absorb them.

Materials that have a low atomic number are good reflectors; in fact, many moderators are also very good reflectors. Human tissues are both good moderators and good reflectors.

(6) Enrichment. Naturally occurring uranium is mostly nonfissile ^{238}U and less than 1% fissile ^{235}U. If the world's entire supply of natural uranium ore were collected into a giant sphere and covered with drinking water, it would not be critical. However, uranium can be enriched. Uranium is said to be enriched when the percentage of ^{235}U atoms has been increased above the percentage found in natural uranium. As the enrichment increases, the number of fissile atoms that can capture neutrons and then undergo fission increases, and fewer nonfissile atoms are available to capture neutrons and prevent the fissioning process. Therefore, the greater the level of enrichment, the easier it is for an accumulation of fissile material to attain criticality.

(7) Interaction. The escape of neutrons from one accumulation or "pile" of fissile material, and their subsequent entrance into another accumulation that can cause more fissions, is called interaction. Interaction can occur if accumulations of fissile material are close enough together. For this reason, accumulations of fissile material must be stored far enough apart to prevent interaction. Keeping accumulations of fissile material at preestablished distances apart is a commonly used criticality control technique.

(8) Type of Material. Each type of fissile material has different nuclear properties. For example, the amount of ^{235}U needed to support a chain reaction is about 1 kg, whereas the amount of ^{239}Pu needed is only about 1/2 kg.

(9) Nuclear Poisons. Nuclear poisons are materials that absorb neutrons without undergoing fission. This absorption decreases the number of neutrons available to cause a fission. Examples of nuclear poisons include cadmium, boron, and samarium.

The nine factors mentioned above can interact and make the problem of determining safe handling procedures for fissile materials very complex. Because of this complexity, a criticality safety expert should be consulted whenever questions arise concerning criticality safety.

REFERENCES

Brodsky, A. 1965. "Determining Industrial Hygiene Requirements for Installations Using Radioactive Materials." Amer. Ind. Hyg. Ass. J. 26:294-310.

International Commission on Radiation Units and Measurements (ICRU). 1980. Radiation Quantities and Units. ICRU 33, Washington, D.C.

Interagency Task Force on Health Effects of Ionizing Radiation. 1979. Report of the Interagency Task Force on the Health Effects of Ionizing Radiation. U.S. Department of Health, Education and Welfare, Washington, D.C.

Lederer, C. M., and V. S. Shirley, eds. 1978. Table of Isotopes. John Wiley and Sons, New York.

National Council on Radiation Protection and Measurements (NCRP). 1971. Basic Radiation Protection Criteria. NCRP 39, Washington, D.C.

Radiological Health Handbook. 1970. U.S. Department of Health, Education and Welfare, Bureau of Radiological Health. Available from U.S. Government Printing Office, Washington, D.C.

U.S. Department of the Army and Defense Logistics Agency. Medical Services - Control and Recording Procedures for Exposure to Ionizing Radiation and Radioactive Materials. AR 40-14, DLAR 1000.28, Washington, D.C.

APPENDIX A

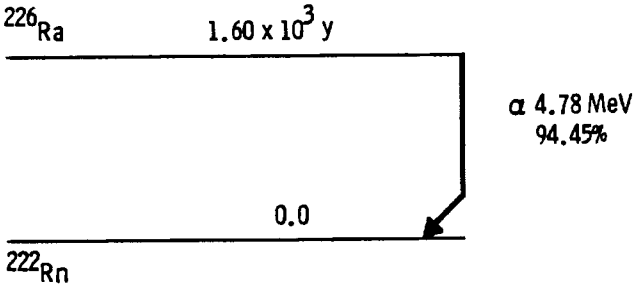
DECAY SCHEMES

APPENDIX A

DECAY SCHEMES

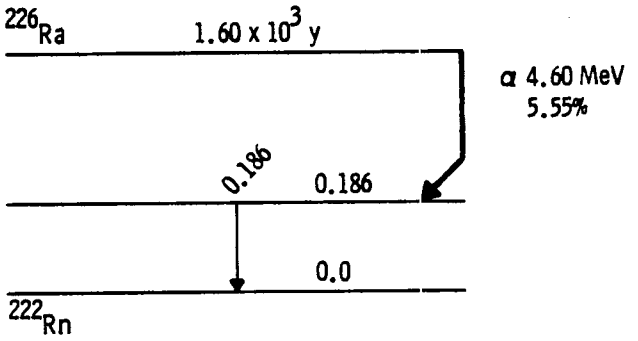
Decay schemes are diagrammatic representations of radioactive decay pathways. The chemical symbol, mass number, and half-life of the parent nuclide appear on the uppermost horizontal line. Decay leading to an increase in the N/Z ratio (alpha emission, positron emission, and electron capture) is indicated by a bent arrow leading to the lower left; decay leading to a decrease in the N/Z ratio (electron emission) is drawn with an arrow leading to the lower right. These arrows terminate on horizontal lines that represent the nuclear energy levels of the daughter nucleus. If the daughter nucleus formed is in an excited state, then gamma rays are emitted until the ground, or unexcited, state is reached. Gamma rays are represented by vertical lines that may be either straight or wavy. The maximum kinetic energy of the emitted particle or the energy of the gamma ray is indicated near the appropriate arrow. If more than one pathway may be followed, the fractional or percentage occurrence of each pathway is indicated.

As mentioned in Section 1.2.2, radium-226 (which has a half-life of 1600 years, or 1.60×10^3 years) can undergo radioactive decay by five pathways. The first and most common pathway consists of the emission of an alpha particle that has 4.78 MeV of kinetic energy. In this case, the daughter nucleus, radon-222, is not in an excited state, so no gamma ray is emitted. A decay scheme showing this pathway is shown below:



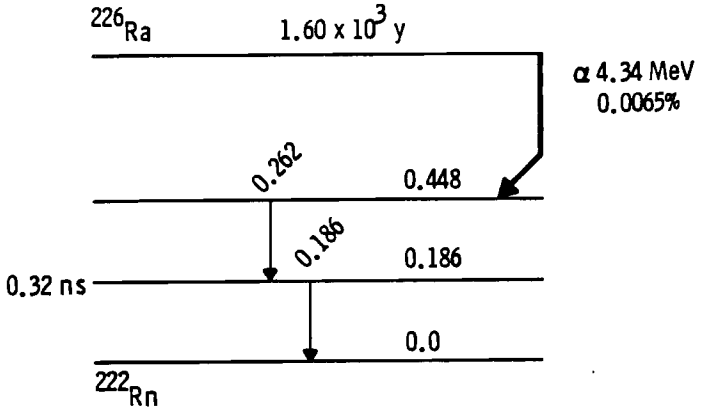
This scheme indicates that of all the decays of ^{226}Ra to ^{222}Rn, 94.45% proceed by the emission of only a 4.78-MeV alpha particle.

The next most common pathway for the decay of ^{226}Ra is the emission of an alpha particle that has a kinetic energy of 4.60 MeV. The daughter nucleus, ^{222}Rn, is in an excited state, and a gamma ray of 0.186 MeV is emitted. The scheme for this pathway is shown below.

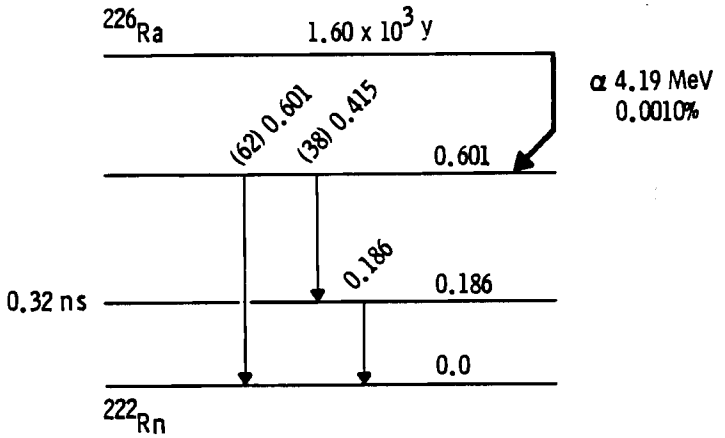


This scheme shows that, of all the decays of ^{226}Ra, 5.55% decay by this pathway. The numbers to the right of the horizontal lines represent the energy level of the daughter nucleus, in MeV. The straight vertical line between the 0.186 line and the 0.0 line represents a gamma ray that has an energy of 0.186 MeV (shown by the numbers above the gamma ray).

A third pathway by which ^{226}Ra decays is the emission of a 4.34-MeV alpha particle. The ^{222}Rn daughter nucleus is left in an excited state and loses energy by the emission of two gamma rays, one that has 0.262 MeV and a second that has 0.186 MeV of energy. The two gamma rays are emitted in quick succession, the 0.262-MeV gamma first, followed by the 0.186-MeV gamma. The 0.186-MeV level of ^{222}Rn has a half-life of 0.32 nsec (3.2×10^{-10} sec). This amount of time is not long enough for this energy level to be considered an isomer of ^{222}Rn. The decay scheme is shown on the next page.

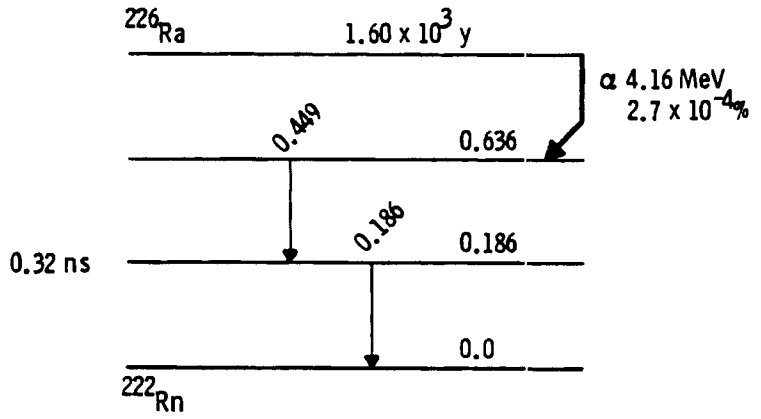


The fourth pathway by which ^{226}Ra decays is the emission of an alpha particle that has 4.19 MeV of kinetic energy. The ^{222}Rn daughter nucleus is in an excited state and releases its extra energy in two ways: 62% of the time, a single gamma ray with 0.601 MeV of energy is emitted; 38% of the time, two gamma rays are emitted, one following the other in a cascade. The total energy of the two gamma rays (0.415 and 0.186 MeV) is equal to the energy of the single emitted gamma ray. The scheme may be drawn as follows:

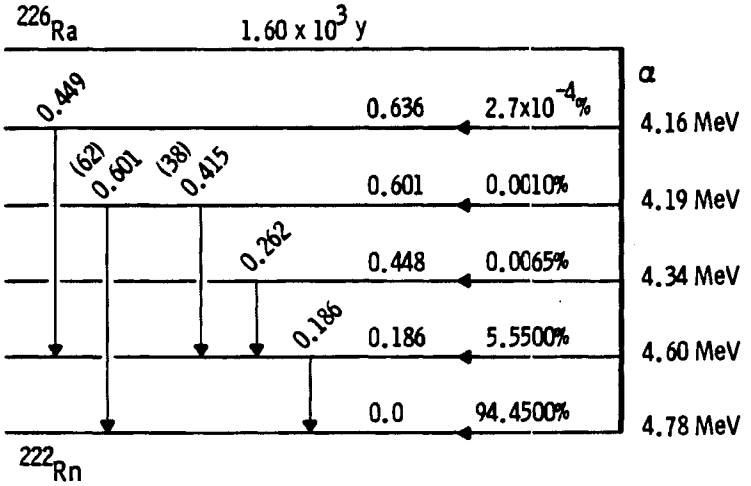


The (62) and (38) preceding the 0.601-MeV-level gamma rays indicate the percent of gamma rays following each pathway.

The final decay pathway that ^{226}Ra can follow is the emission of an alpha particle that has a kinetic energy of 4.16 MeV. The pathway is followed infrequently, by only 2.7×10^{-4}% of all decays. The ^{222}Rn daughter nucleus is an excited state and releases 0.636 MeV of energy in the form of two gamma rays of 0.449 and 0.186 MeV. These gamma rays are emitted in a cascade. The decay scheme for this pathway can be drawn as follows:

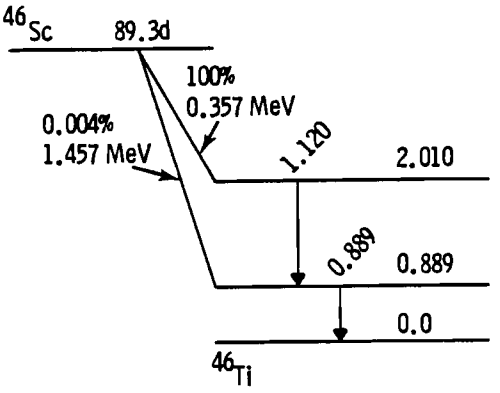


When a radionuclide follows multiple decay pathways, as does ^{226}Ra, the various pathways are all shown on one diagram rather than as several different schemes. When the complete decay scheme is complex, the method of drawing the arrows is altered to save space. Thus, the complete decay scheme for ^{226}Ra would be drawn as below:

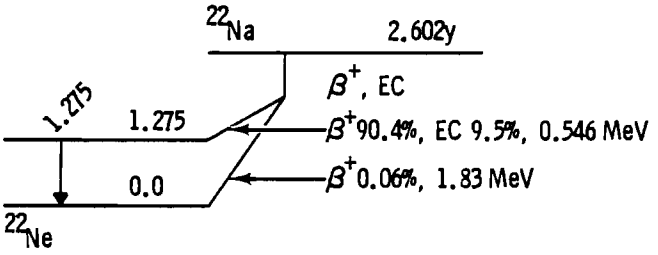


This diagram is one way of presenting the decay scheme information. Notice that the gamma ray resulting from the de-excitation of the 0.186-MeV nuclear energy level follows the 4.60-MeV alpha, the 4.34-MeV alpha, the 4.19-MeV alpha, and the 4.16-MeV alpha.

A decay scheme for beta negative emission, in the decay of scandium-46 to titanium-46, is shown below:



The decay of sodium-22, a positron emitter, to neon-22 is drawn below:



Each publisher has a slightly different method of presenting the data. For example, some publishers number the gamma rays and list the frequency of occurrence and energy in a separate table. The publisher may identify the alpha emissions by the nuclear energy level of the emission (e.g., the 0.635-MeV energy level or the 0.6007-MeV energy level) and may list the information concerning frequency of occurrence and energy in a separate table. In other words, when consulting a decay scheme, the reader should be familiar with the method of presentation used by the publishers.

Information on decay schemes can also be found in tabular form in several references, including the Table of Isotopes (Lederer and Shirley 1978) and the Radiological Health Handbook.

CHAPTER 2. RADIATION INSTRUMENTATION

FIGURES

TABLES

CHAPTER 2. RADIATION INSTRUMENTATION

Ionizing radiation cannot be detected by unaided human senses; instrumentation must be used to detect and measure it. This chapter describes the fundamental characteristics of radiation detection and measurement instruments and their principles of operation, their application, and their limitations. Included in the chapter are an introduction to measurement concepts; a review of instruments used in the field of radiation protection and how they work; information on the calibration of instruments; factors that affect the selection of radiation-monitoring instruments; the types of monitoring instruments and personnel dosimeters available for use; and a brief discussion of statistics and error determination.

Section 2.1 BASIC CONCEPTS IN RADIATION DETECTION AND MEASUREMENT

Numerous types of instruments are used for a wide variety of purposes in the field of radiation protection. Some instruments simply detect the presence of radiation; others give a quantitative measurement of the dose rate or exposure rate produced by the radiation.

Detection and measurement instruments have two basic components, a sensing element and an indicating element. The sensing element, called the detector, responds to the radiation and through various means provides a measurable signal to the indicating element. Common types of indicating elements include meters, recorders, counting scalers, and speakers. Intermediate electronic circuitry may be used to amplify the signal from the detector so that it can be more readily observed in the indicating element.

2.1.1 Characteristics of Instruments

Instruments can be characterized by how radiation interacts with the detector. Several instruments depend for their operation on the ionization of matter by radiation. Other detection systems depend largely upon the excitation of electrons rather than on ionization. Both ionization and excitation

result either directly or indirectly in the formation of electrical charges within the sensitive volume of the detector or its associated circuitry. If an electric field is applied across the sensitive volume of the detector, the electrical charges can be collected because negative charges travel to the positive pole of the electric field and positive charges travel to the negative pole. The collection of the electrical charges causes a build-up of charge that flows through an external circuit.

A second way of distinguishing types of instruments is by whether the flow of charge is recorded as a pulse or a current. An instrument that operates in the pulse mode records an output pulse for each individual interaction between the detector and a particle or photon of radiation. An instrument that operates in the current mode records an average of many individual interactions and subsequent pulse fluctuations. An advantage of the pulse mode is that, for many instruments, the amplitude (size or height) of each individual pulse carries valuable information about the type and energy of the radiation that caused the pulse; in the current mode, information on individual pulses, and thus on individual radiations, is lost. Pulse detectors also have a greater sensitivity than detectors that operate in the current mode; that is, they detect more of the incoming radiation. Because of these advantages, the pulse mode is more commonly used for radiation detection instruments.

A third distinction among instruments is how those that operate in the pulse mode record the pulses. Rate meters record a pulse rate, with readouts in counts per minute (cpm), mR/hr, mrem/hr, etc. Integrating instruments have a digital counting accessory that tallies the pulses for the duration of the measurement, with readouts given in counts, mR, mrad, etc.

Counting devices that accept pulses may have fixed or variable discriminators. The pulse amplitude must be of a certain size to pass the discriminator level and be counted; otherwise it is rejected. If the discriminator level can be varied, information can be obtained about the amplitude distribution of the pulses, and therefore about the types and energies of the radiations.

In nearly all detector systems, a minimum amount of time is required between two interactions in order for them to be registered as two separate

pulses. This interval is called the dead time of the system. Immediately after a pulse, the detector is insensitive to radiation and is unable to respond to other ionizing events. If an ionizing event occurs during this time, it does not produce a pulse. The important consequence of dead time is that a detector in a high-radiation field may indicate less radiation than is actually present. Counts that are recorded can be corrected for dead-time losses, and many laboratory counters have a meter that indicates the percentage of time the counter is dead.

The object of many applications of radiation detectors is to identify the energy distribution of the incident radiation. The ability of a detection system to distinguish between or separate two pulses of slightly different sizes is called its energy resolution. The resolution capabilities of various instruments are discussed later in this chapter.

If a detector counts every particle or photon of radiation that enters its sensitive volume, it has a counting efficiency of 100%. Practically speaking, however, counting efficiencies of 100% are rarely achieved. It is always possible, especially with gamma rays and neutrons, that some radiation will pass through the detector without interacting with it. In order to relate the number of pulses counted to the actual number of radiations incident on the detector, the detector's counting efficiency must be calculated. The absolute efficiency is calculated using Equation (2.1).

$$\text{absolute efficiency} = \frac{\text{number of pulses recorded}}{\text{number of radiations emitted by source}} \qquad (2.1)$$

If the absolute efficiency of the detection system is known and a given number of pulses are recorded for a given time, this equation can be used to determine the activity of the radioactive source (i.e., the number of radiations emitted per unit time).

Another type of efficiency accounts for the fact that all of the radiations emitted by the source may not reach the detector. The intrinsic efficiency of the detector is calculated using Equation (2.2).

$$\text{intrinsic efficiency} = \frac{\text{number of pulses recorded}}{\text{number of radiations incident on detector}} \qquad (2.2)$$

2.1.2 Source Characteristics

In the detection and measurement of radiation, consideration must also be given to characteristics of the radioactive source. The emission of radiation from a radioactive source is generally assumed to be isotropic; that is, radiation is emitted by the source in all directions with equal intensity. In order for all of the radiation emitted by the source to be detected, the source must be completely enclosed within the sensitive volume of the detector. This type of counting arrangement is called 4π geometry (because the solid angle subtended by the detector at the source position is 4π steradians). Most detection systems do not achieve 4π geometry because the source is placed outside the detector and only a fraction of the emitted radiation is directed toward the sensitive volume. The geometry factor is the fraction of the source sphere that actually intercepts the detector. It can be used to determine the actual number of radiations being emitted by the source.

Other source factors that must be considered are self-absorption, radiation attenuation, and the inverse-square law. When a radioactive source produces radiation, there is a finite probability that the radiation will lose its energy within the source itself. This process, called self-absorption, occurs most frequently with encapsulated alpha and beta sources because the energy of the particles is absorbed by the capsule material. Radiation may also lose its energy in the air between the source and the detector, or in the shielding of the detector before it reaches the sensitive volume, and this attenuation must be considered when attempting to determine the activity of the source. Finally, assuming that the radioactive source is a point source (very small compared to the distance to the detector) and that particles or photons radiate outward from it, the number of radiations in a unit area falls off with distance. The greater the distance between the source and the detector, the fewer the radiations entering the sensitive volume of the detector and therefore the lower the count rate. A complete discussion of this principle, called the inverse-square law, is presented in Chapter 6.

Section 2.2 RADIATION PROTECTION INSTRUMENTS AND HOW THEY WORK

Instruments used for radiation protection are of three general types: gas ionization detectors, scintillation detectors, and semiconductor detectors. The principles on which these detectors work and the types of detectors in each group are described in this section.

2.2.1 Gas Ionization Detectors

As radiation passes through a gas, it gives energy to orbital electrons, causing ionization and excitation of the gas atoms through the mechanisms described in Chapter 1. Gas ionization detectors use the process of ionization to detect the presence of radiation.

A. Principles of Operation. A simplified diagram of a gas ionization detector is shown in Figure 2.1. The detector assembly usually consists of a power supply and a closed, electrically conductive cylinder or chamber that is filled with gas. The chamber walls are usually made of metal, which can be

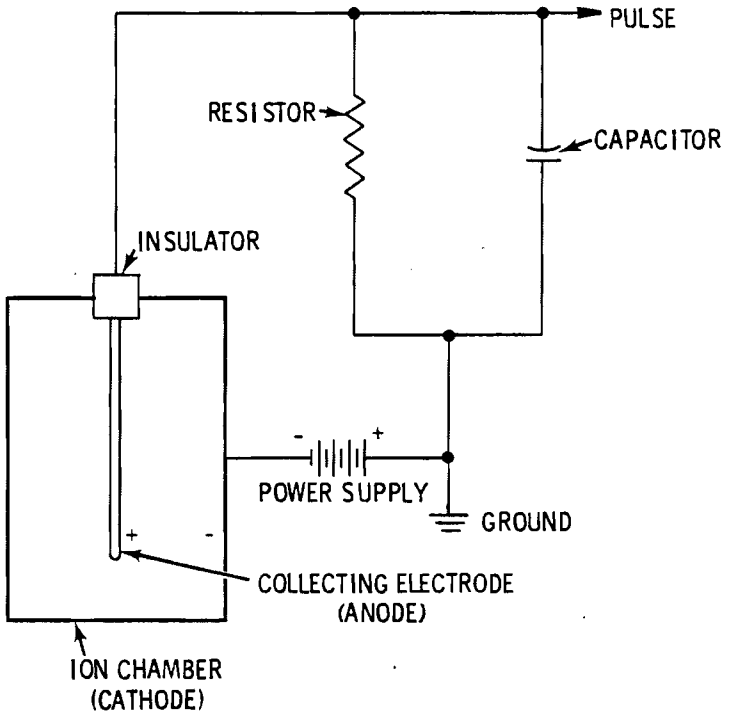


FIGURE 2.1. Simplified Version of a Chamber Used to Collect Ions

penetrated by photons and some high-energy beta particles. The chamber may have a "window" made of a material such as mylar, which can be easily penetrated by alpha particles and lower-energy beta particles.

The positive and negative poles of the power supply are called electrodes. A thin wire in the center of the chamber is connected to the positive electrode of the power supply and is called the central collecting electrode, or anode. The chamber wall is connected to the negative electrode of the power supply and is called the cathode. As radiation passes through the gas that fills the chamber, it gives energy to the orbital electrons of the gas atoms and may cause them to be removed from the originally neutral gas atoms. This ionization process results in the formation of a free electron (negative ion) and a positive gas atom (positive ion), which together are called an ion pair. Repeated interactions between radiation and the fill gas in a closed chamber gradually cause the degradation of the gas until eventually the detector loses its effectiveness, and either the degraded gas is removed from the chamber and replaced with new gas, or the entire detector is replaced.

The number of ion pairs created in a given volume of the chamber's fill gas depends on the type of gas used and the type and energy of the radiation. A dense gas has more atoms for the radiation to interact with than does a less dense gas and thus leads to the creation of more ion pairs. Alpha particles, which are relatively heavy and slow and have a double positive charge, create many ion pairs within a very short distance as they travel through the fill gas. They typically give up all of their energy to the gas within a few centimeters. Beta particles, which are much smaller and faster than alpha particles, do not interact as readily with the orbital electrons and thus create fewer ion pairs. Gamma rays and x rays, which are uncharged and have negligible mass, interact indirectly with the gas (see Chapter 1) and produce even fewer ion pairs. If an alpha particle, a beta particle, and a gamma ray with identical energies passed through the same volume of a fill gas, the alpha particle would create tens of thousands of ion pairs, the beta particle a few hundred ion pairs, and the gamma ray just a few ion pairs per centimeter of gas. The number of ion pairs created also depends on the energy of the radiation. On the average, one ion pair is produced for every 30 to 35 eV of energy transferred

to the gas. Thus, a single 1-MeV radiation that loses all of its energy in a gas creates approximately 30,000 ion pairs; a 2-MeV particle creates 60,000 ion pairs.

When voltage is applied across the chamber, the ion pairs produced in the gas by the incident radiation move toward their respective electrodes: the negatively charged electrons move rapidly to the positively charged anode, and the positively charged ions, which are much heavier, move very slowly toward the negatively charged chamber wall. The electrons that collect on the anode produce a build-up of charge. The collected charge flows through the external circuit as a pulse or surge of current. Each pulse represents the interaction of one particle or photon of radiation with the gas. The pulses flowing through the external circuit of the instrument can be recorded in one of two ways, depending on the type of electronic circuit used. If a nonintegrating, or differential, circuit is used, each individual pulse can be tallied, which gives a record of the total number of ionizing radiations entering the chamber; if an integrating circuit is used, the total current flow over a given period of time can be measured. The total current flow is proportional to the degree of ionization in the chamber.

The magnitude of the voltage applied to the electrodes is another factor that affects the number of electrons collected on the anode and the resulting charge. Figure 2.2 shows the relationship between the applied voltage and the pulse height in the circuit. In this figure, six regions can be observed: 1) the recombination region, 2) the ionization chamber region, 3) the proportional region, 4) the limited-proportional region, 5) the Geiger region, and 6) the continuous-discharge region.

(1) Recombination Region. In this region, the voltage across the electrodes is relatively low, and the force of attraction between the ions and the electrodes is not great. Therefore, most of the positive and negative ions produced by the radiation are attracted to each other, rather than to the electrodes, and they recombine. As the voltage applied to the electrodes is increased, fewer ions recombine. However, no radiation detectors operate in this region.

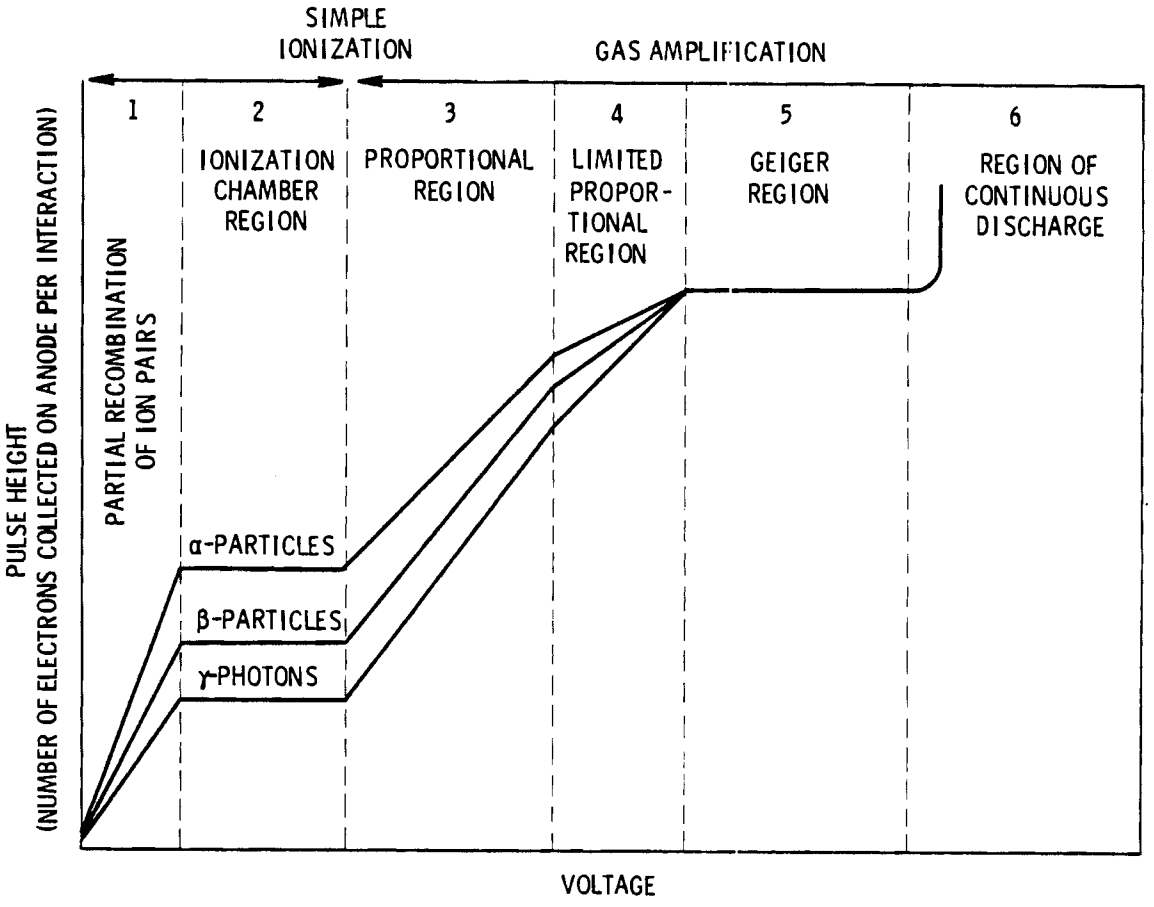


FIGURE 2.2. Relationship Between Applied Voltage and the Number of Electrons Collected on the Anode

(2) Ionization Chamber Region. At a certain voltage, the force of attraction between the ions and the electrodes is sufficient to cause all of the electrons produced by the incident radiation to be collected on the anode. Subsequent moderate increases in the voltage do not create any further increase in the electron current: a saturation voltage has been reached. (For this reason, the ionization chamber region is also called the saturation region.) The number of electrons collected at the anode is a function of the amount of ionization occurring in the chamber.

Figure 2.2 shows three curves in the ionization chamber region, one each for alpha particles, beta particles, and gamma rays (photons). Because alpha particles create a larger number of ion pairs per path length than the other

radiations do, more electrons are collected on the anode and a larger pulse is produced in the external circuit. The pulse height for beta particles, which create fewer ion pairs than alphas, is slightly smaller, and the pulse height for gamma rays is the smallest. Thus, in the ionization chamber region, the different types of radiations can be distinguished from each other because of the different pulse heights produced in the external circuit.

(3) Proportional Region. If the voltage between the anode and the cathode is increased, the number of ion pairs collected is larger than the number of primary ion pairs (those initially formed by the incident radiation). At high voltages, the primary negative ions (i.e., electrons) are accelerated toward the anode fast enough to cause additional ionization of the gas, creating secondary ion pairs. The secondary electrons that are then accelerated toward the anode may also have enough energy to cause even further ionization of the gas. This multiplication or avalanche of electrons moving toward the anode is called gas amplification, and in the proportional region the avalanche is restricted to the vicinity of the primary ionizations. The gas amplification factor, or multiplication factor, is a measure of the number of secondary electrons produced by one primary electron. Thus, if one primary electron causes 10,000 secondary electrons to be produced, the multiplication factor is 10,000. (In the ionization chamber region, the multiplication factor is 1 because the relatively low voltage across the electrodes does not result in an avalanche, or multiplication effect.) In the proportional region, the total number of ion pairs eventually formed is proportional to the number of primary ion pairs formed by the incident radiation, and the multiplication factor is constant over small voltage ranges within the region. Detectors operating in the proportional region have multiplication factors up to 10^6, depending on the applied voltage, but the typical factor is 10^4. These detectors, like those operating in the ionization chamber region, can distinguish between alpha, beta, and gamma radiations.

(4) Limited Proportional Region. At the upper range of the proportional region, the gas amplification factor is no longer constant for a given voltage range but can change markedly with small changes in the applied voltage. This region is called the limited-proportional region and, in general, has no useful purpose for radiation measurement.

(5) Geiger Region. A further increase in voltage leads to the Geiger region. The gas amplification in this region is so extensive that an avalanche of electrons spreads along the entire length of the instrument's anode, and all pulses are the same size, regardless of the type of radiation that initiated the ionization. Thus, a detector operated in the Geiger region cannot distinguish between the different types of radiation. The pulses in the Geiger region are much larger than those in any of the previous regions. In fact, the production of only one primary ion pair results in an easily measurable pulse ($\sim$1 V).

As positive ions approach the cathode wall of the detector, they have so much energy (because of the high voltage in the Geiger region) that they attract electrons from the wall and become neutral atoms. During this process, a low-energy x ray is often emitted that can cause further ionization. If this additional ionization were allowed to proceed, the detector would remain in a continual state of discharge and would not count a second pulse. To terminate, or quench, the perpetual ionization in the detector, a small amount of quenching gas is added to the chamber. The quenching gas transfers its electrons to the positive ions, and the electrons and positive ions combine to create neutral gas atoms. Without its electrons, the quenching gas has a positive charge; it migrates to the cathode and collects electrons to become neutralized. The energy produced in this process goes into the dissociation of the gas molecule rather than the production of an x ray. Bromine, chlorine, ethanol, and methane are typically used as quenching gases.

(6) Continuous-Discharge Region. If the voltage is increased still further, arcing occurs across the electrodes, and pulses are registered continuously even if no radiation is present. Instruments operated in this region can be permanently damaged in a short time.

The three types of ionization instruments commonly used by radiation protection personnel--ionization chambers, proportional counters, and Geiger-Mueller counters--correspond to the three regions of the pulse height-voltage curve in which radiations can be detected.

B. Ionization Chambers. Instruments designed to operate in the ionization chamber region of Figure 2.2 are called ionization chambers, or ion chambers. They can be passive or active detectors.

(1) Passive Ion Chambers. In a passive ion chamber, a voltage is placed across the electrodes in a process called charging. The chamber is then separated from the charger and placed in a radiation field. The ions formed by the incident radiation neutralize the charge, and the subsequent drop in voltage can be measured and correlated to the amount of radiation that was present. Two types of passive ion chambers are pocket ionization chambers and condenser chambers.

Pocket ionization chambers, also called pencil dosimeters, are integrating instruments that record the total current flow, or true charge, produced by the radiation entering the chamber. These dosimeters have a metal-coated quartz fiber that is attached at one end to a rigid metal electrode and suspended in a small gas-filled chamber. When a positive charge is placed on the electrode, the charge is also transferred to the fiber, and because like charges repel, the fiber moves away from the electrode. When radiation ionizes the fill gas in the chamber, the resulting negatively-charged electrons combine with and neutralize some of the positive charges on the fiber and electrode (the fiber and electrode are said to discharge). This results in a decrease in voltage between the two, and the fiber moves closer to the electrode. How far it moves depends on the number of electrons formed by the radiation; thus, the distance between the electrode and the fiber indicates how much radiation the dosimeter was exposed to.

Self-reading pencil dosimeters are equipped with a built-in microscope and a scale that enables the wearer to read the exposure at any time. When the dosimeter is fully charged, the fiber lies on the zero point on the scale. As the fiber discharges in response to ionizing radiations, it moves along the scale. Non-self-reading pencil dosimeters must be inserted into a specially designed voltmeter to be read. If a dosimeter is dropped or subjected to other sudden motions, it may discharge and incorrectly indicate a very high exposure.

Another type of passive ion chamber, the condenser chamber or condenser R-meter, is used to make highly accurate and precise measurements. Condenser chambers are similar to non-self-reading pocket ionization chambers but are very carefully constructed and have walls of uniform thickness so that the

energies of incident photons can be measured. These instruments also respond to beta rays with energies higher than 1 MeV. If the inside of a condenser chamber is coated with boron, it also responds to thermal (low-energy) neutrons.

(2) Active Ion Chambers. Active ion chambers have a built-in voltage source. The circuits in these chambers can be nonintegrating, registering a pulse for each particle or photon of radiation that interacts with the fill gas, or integrating, measuring the total current produced by the ionizations.

The most popular use of active ionization chambers is as portable instruments to survey for beta and gamma radiation. These instruments come in various forms, shapes, and sizes, but the most common type is the pistol-shaped, portable rate meter known as the "Cutie Pie." Most of these survey instruments are thin-window ionization chambers that have a removable shield over the window end of a cylindrical chamber. When the shield is removed, the instrument responds to both beta and gamma radiations, but when the shield is in place, only the gamma rays can penetrate it to enter the chamber. Therefore, to get a correct beta reading, it is necessary to take two readings, one with the shield on and one with it off. The shield-on reading is then subtracted from the shield-off reading to give the beta reading.

Active ion chambers can also be used to measure alpha particles. A chamber for this purpose is usually designed so that the alpha source can be placed inside the cylindrical chamber. Because the chamber completely surrounds the source, which is emitting particles uniformly in all directions, all of the alpha particles emitted from the source deposit their energy within the chamber. This type of counting system is an illustration of 4π geometry and results in a near-100% counting efficiency.

C. Proportional Counters. A proportional counter is a gas ionization detector that is operated in the proportional region of the pulse height-voltage curve (see Figure 2.2). The anode, or collecting electrode, is a loop of very thin wire (approximately 0.025 mm) that is usually made of fine, clean tungsten with minimal surface irregularities. The cathode, or outer sheath of

the cylindrical chamber, is either metallic or metal- or carbon-coated glass. Detectors operating in the proportional region can have either nonintegrating or integrating circuits.

A mixture of 10% methane and 90% argon, known as P-10 gas, is commonly used as the fill gas in proportional counters. A mixture of 4% isobutylene and 96% helium can also be used. These gases provide stable operation and high gas amplification. Air is rarely used as the fill gas because oxygen easily captures electrons before they reach the anode, reducing gas amplification.

The proportional counters used today are either gas-flow or sealed. In gas-flow proportional counters, gas flows through the counting chamber at a very low rate, removing the degraded gas and any contaminants. Because of the continual replacement of the fill gas, these detectors have a long life. Sealed proportional counters have a finite life because the fill gas, which is sealed inside the counting chamber, degrades over time as incoming radiations interact with it. However, the chamber can be emptied and completely refilled with new counting gas.

(1) Gas-Flow Proportional Counters. Before a gas-flow proportional counter is operated, residual air and contaminants must be removed with a brief, large flow of counting gas. This process is called purging. The chamber of a simple gas-flow proportional chamber is hemispherical or sometimes cylindrical. The radiation source is typically positioned at the bottom of a hemispherical chamber or in the middle of a cylindrical chamber. If the source is suspended in the chamber, 4π geometry is achieved. If the source is positioned at the bottom of the chamber, the device is referred to as a 2π counter.

Windowless gas-flow counters are used to count alpha and beta particles. Because alpha particles have a much higher specific ionization than beta particles (they form many more ion pairs per path length as they move through the fill gas), the large pulses of electronic charge that result from alpha interactions with the fill gas can be electronically distinguished from the smaller beta pulses by adjusting the operating voltage. If the count rate versus the operating voltage is plotted, two plateaus are observed (see

Figure 2.3). At low voltages, only the alpha particles produce pulses because they are more energetic and more highly ionizing than the beta particles. This portion of the curve is called the alpha plateau. If the applied voltage is increased past the alpha plateau, the counting rate begins to increase as gas amplification is caused by increasing numbers of beta particles. After a transition region, another plateau is reached that represents the pulse created by alpha and beta particles together. This plateau is often referred to as the beta plateau. Because beta particles vary widely in their energies, the beta plateau is not as flat as the alpha plateau.

Alpha particles on surfaces can be detected using a specially designed gas-flow proportional counter. The detector is flat and has a window made of aluminized mylar. The counting gas is frequently propane, which is attached to the counter in small, interchangeable metal bottles. This survey instrument is especially useful in areas where alpha surveys are required and gamma radiation levels are high (50 to 500 mR/hr), because it can discriminate against the smaller pulses produced by gamma rays.

(2) Sealed Proportional Counters. A specially designed sealed proportional counter can be used to detect and measure low-energy (thermal) neutrons. Neutrons do not interact directly with the orbital electrons of the

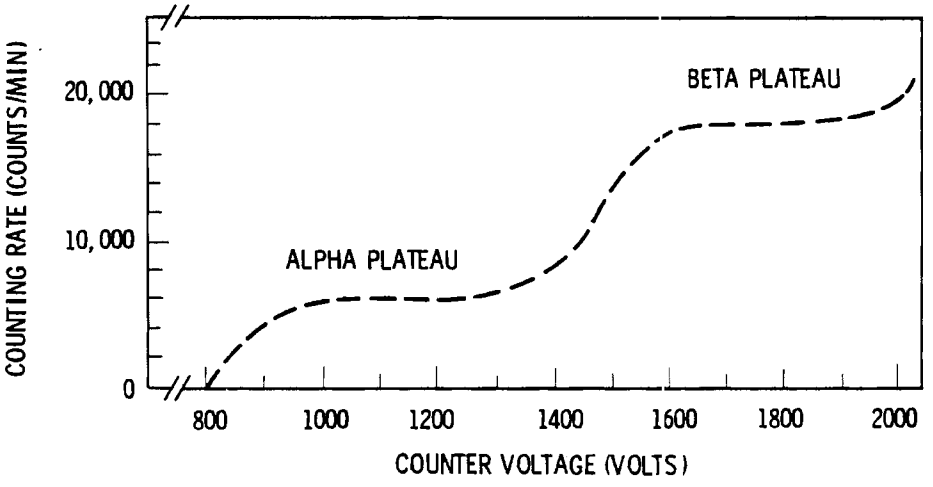


FIGURE 2.3. Plateaus for Typical Proportional Counter

fill gas (they are not directly ionizing radiation--see Chapter 1). Therefore, the detection and measurement of neutrons relies on the interaction of the neutrons with some material capable of causing ionizations. The boron trifluoride (BF_3) gas proportional counter is the most commonly used instrument for this purpose. Low-energy neutrons interact with the boron to form alpha particles, which can then ionize the gas. The BF_3 counter can also be used to measure high-energy (fast) neutrons. For this, the chamber is wrapped in polyethylene, paraffin, or some other hydrogen-containing material that slows down (reduces the energy of) the incident neutrons. These instruments, often referred to as rem meters, have the advantage of being insensitive to most other types of radiation. The small pulses produced by gamma rays can be discriminated out electronically.

D. Geiger-Mueller Counters. Geiger-Mueller (GM) counters are gas ionization detectors designed to operate in the Geiger region of the curve in Figure 2.2. They can be used as pulse counters in the laboratory or as portable survey instruments to detect alpha, beta, and gamma radiation. However, they cannot be used to distinguish between the different types of radiation because all of the pulses produced in the Geiger region of the pulse height-voltage curve are the same size.

The detector itself is a stainless-steel tube that contains the fill gas (usually argon) and the anode and that may have an end or side window. Pulses are electronically transmitted to a counter or a meter, and the readout is generally given in cpm. Some GM instruments are designed to read out in mR/hr to R/hr in response to gamma rays with energies between 60 keV and 1.5 MeV. However, these instruments should not be used as dose rate or exposure rate meters because they produce pulses of the same size regardless of the energy of the phtons causing the ionization. True dose rate meters give a response that is related to the energy of the photons.

Wall and window thicknesses, which are expressed in mg/cm^2,(a) range from 30 mg/cm^2 (for counting gamma rays and high-energy beta particles) down to 0.4 to 1.4 mg/cm^2 (for counting alpha particles and low-energy beta

(a) Thickness (mg/cm^2) = density of the material (mg/cm^3) x linear thickness (cm).

particles). One of the more popular GM survey meters uses a tube (10 cm long and 2 cm in diameter) encased in a stainless-steel housing that contains a window. The window can be opened to admit beta particles and gamma rays, or closed to admit only gamma rays. Thus, the beta contribution to the radiation field can be determined with this instrument.

For monitoring alpha and beta radiation, a "pancake" GM tube is used. The detector is a flat, round cylinder with a large window that is approximately 5 cm in diameter and 16 cm^2 in total areas. The thickness of the window is 1.4 to 2.0 mg/cm^2. The detector is sensitive to alpha radiation with energies above 3 MeV and to beta radiation with energies above 40 keV. In addition, the detector has a shield (usually made of tungsten) over all surfaces except at the window location, to reduce the influence of gamma radiation. To protect the thin window, a wire screen is sometimes provided.

Portable GM survey meters can be equipped with either a head set containing ear phones, or a speaker attached to the instrument case. Each time a pulse is recorded in the counting circuit, a click is heard. These devices are extremely useful in surveying for radiation because their response is much faster than the meter indication. The audible circuit is separate from the meter circuit and does not fail even if the device saturates and the meter indicates zero.

Geiger-Mueller counters are probably the most widely used and versatile instruments for detecting radiation. They are inexpensive, easy to operate, sensitive, and reliable. However, their use in or near very high radiation fields requires caution because most counters saturate in such a field. The incident radiation enters the sensitive volume of the tube at such a rate that the tube is in a state of continuous discharge, and the count rate circuit fails to function properly. As a result, the meter begins to respond but then falls off and reads near zero rather than off the high end of the scale. A person entering a very high radiation area might not realize it because the GM had failed.

2.2.2 Scintillation Detectors

Shortly after x rays were discovered, researchers found that certain materials fluoresce, or emit visible light, when struck by radiation. These

materials are referred to as phosphors, or scintillators. Scintillation detectors were among the earliest instruments for detecting and measuring ionizing radiation and they are still widely used today.

A. Principles of Operation. As radiation enters and passes through a phosphor, it gives up its energy to electrons in the phosphor by both ionization and excitation. Excited electrons move into defects, or gaps, in the atomic structure of the phosphor, called traps. When the electrons escape from the traps to return to lower energy levels, the excess energy is released in the form of visible light. This process is called scintillation, and the light flashes produced are called scintillations.

The light flashes generated in the phosphor can be detected and related to the incident radiation by means of a photomultiplier tube, which is a combination of a photocathode and an electron multiplier. A photocathode converts flashes of light (light photons) into electrons by the photoelectric effect (see Chapter 1). An electron multiplier multiplies the number of electrons using a series of electrodes, called dynodes, which are positively charged. The electrons from the photocathode are accelerated to the first dynode by the application of enough voltage to cause multiple emission of secondary electrons at the first dynode. The secondary electrons are then accelerated to subsequent dynodes, resulting in further multiplications. The typical voltage between each multiplying stage is 50 to 250 V, with each dynode having a more positive voltage than the preceding one. After the last multiplying stage, the electrons are collected at the anode of the photomultiplier tube and fed to an external circuit in the form of a pulse. Photomultiplier tubes typically have a gain, or multiplication, of 10^6. That is, the number of electrons released by the photocathode is multiplied a million times by the time all of the electrons reach the last dynode.

The output current from the photomultiplier tube is then detected and analyzed by the electronic circuit. The extent of the electronic circuitry depends upon the application of the system. A simple circuit, used simply to detect radiation, consists of a battery-operated power supply and an amplifier with a pulse shaper and a rate meter. However, when the device is used for analyzing the energies of the photons emitted by a radioactive material, the

circuit includes a pulse height analyzer, a scaler, and other equipment. A pulse height analyzer sorts the detector signals, or pulses, by size and stores them in appropriate pulse height channels. The size of a detector signal, and thus the channel to which it is assigned, corresponds to the energy of the incident photon.

A single-channel analyzer can analyze only one channel at a time; that is, it can count the number of pulses within a size limit that is manually set (using upper- and lower-level discriminators) on the face of the analyzer. If, for example, the lower-level discriminator is set to reject pulses below 50 kV and the upper-level discriminator is set to reject pulses above 60 kV, only those pulses within the 50- to 60-kV range will be counted. By starting at the lower end of the scale and going upward, an operator can identify which channels have the greatest number of counts, or peaks. Each peak corresponds to photons of a specific energy, which in turn correspond to specific radionuclides. This process is called spectrometry.

A multichannel analyzer has up to several hundred or several thousand single-channel analyzers automatically sorting pulses into specific channels. The data that is accumulated is displayed as a plot with channel number (or photon energy) on the x axis versus the number of counts in a specific channel on the y axis. This plot is called a spectrum. Display modes include oscilloscope screens, x-y plotters, and electric typewriters, which type out channel numbers versus counts. Because each radionuclide has its own distinct spectrum, spectrometry can be used to identify unknown radionuclides.

B. Inorganic Scintillators. Inorganic scintillators are inorganic (not carbon-containing) salts that form regular crystalline lattices. These lattices contain small amounts of impurities that activate the scintillation process (that is, they cause the crystal to emit light when it is exposed to radiation).

Crystals of the alkali halides (e.g., sodium iodide) are the most widely used class of scintillators. Sodium iodide (NaI) is a dense material with which gamma rays interact readily. Crystals of this material are activated for scintillation by the deliberate inclusion of a trace amount of thallium (Tl). These crystals, which can be used to detect gamma and x radiation, can

be produced in a solid cylinder or shaped like a well. The well shape is formed from a crystal with a hole drilled part way into it; small vials or cylindrical samples that are placed in the well are, in effect, surrounded by the crystal, a configuration that results in the detection of most of the emitted radiation.

Sodium iodide crystals are very effective for high-efficiency analysis of gamma-ray spectra. However, these crystals have a relatively poor energy resolution; that is, they cannot easily distinguish between, or separate, photon peaks of slightly different sizes. They are therefore of limited use in distinguishing between radionuclides that emit gamma rays of very similar energies.

Zinc sulfide (ZnS), another inorganic salt, is activated for scintillation by the inclusion of silver (Ag) and is used to detect and measure heavy charged particles, such as alpha particles. A zinc sulfide crystal must be about 20 μm thick in order to detect alpha particles. If the material is thicker or thinner than this, its detection efficiency decreases. In portable alpha survey meters, the zinc sulfide can be applied to the back of a thin window or sometimes painted right on the face of the photomultiplier itself.

When large areas or large volumes of a scintillator are needed, as in whole-body counters, the use of inorganic crystals involves high cost and considerable handling problems because the crystals must be protected from thermal and mechanical shock. These problems can be minimized by the use of organic scintillating materials.

C. Organic Scintillators. Organic scintillators contain carbon, which combines readily with hydrogen and oxygen. These scintillators have a low atomic number and a relatively low density, which makes them suitable for beta counting and, in the case of liquid organics, for alpha counting (the density is too low for high-efficiency counting of gamma rays). Organic scintillators can take the form of solid crystals, liquids, or plastics because the scintillation process arises from a transition in the energy level of a single molecule, and the transition does not depend on the physical state of the scintillator material.

(1) Organic Crystals. The two most common organic crystalline scintillators are anthracene ($C_{14}H_{10}$) and stilbene ($C_{14}H_{14}$). Anthracene has the highest efficiency for light output of any organic scintillator, but both materials are fragile and difficult to obtain in large sizes. They can be used to detect high-energy beta particles, but low-energy betas are either self-absorbed or absorbed by the surroundings before they can interact with the crystal. To overcome this problem, liquid organics can be used.

(2) Liquid Organic Scintillators. Liquid scintillators are made by dissolving an organic scintillator material, called the solute, in an organic solvent. The radioactive source, or sample, is then dissolved in the solution. Because all the radiations emitted by the sample must pass through some portion of the scintillator solution, counting efficiencies can approach 100%. This method is particularly advantageous for counting low-energy beta emitters, such as ^{3}H and ^{14}C, and can also be used for alpha emitters.

The scintillator solution, which is often called a cocktail, consists of the radioactive sample, the organic solvent, a primary scintillator solute (primary fluor), and sometimes a secondary solute (secondary fluor) and a solubilizing agent (diluent). The solvent, which is often toluene, xylene, or dioxane, absorbs most of the energy of the beta particles through particle interactions (see Chapter 1) and transfers it to the primary fluor. The primary fluor is made up of large organic molecules, such as p-terphenyl or PPO (chemical name: 2,5-diphenyloxazole), that scintillate after they have received the excitation energy from the solvent. The concentration of the primary fluor in the cocktail is usually about 1%. The secondary fluor absorbs the light emitted by the primary fluor and re-emits it at a somewhat longer wavelength, which is closer to the wavelength needed for optimum operation of the photomultiplier tube. A diluent such as a hydrocarbon, ether, or alcohol may be added to the cocktail if the radioactive sample does not readily dissolve in the solvent.

Although diluents favorably change the character of the solvent, they also decrease the counting efficiency by interfering with the transmission of light to the photomultiplier, as does the introduction of the radioactive sample itself. This interference, known as quenching, may limit the amount of

a radioactive sample that can be effectively incorporated into the solution. Examples of diluents that are effective but that have a strong quenching action are phenols, amines, aldehydes, and nitro- and iodo-compounds (compounds containing NO_2 or iodine), as well as colored substances. All modern instruments for liquid scintillation counting have electronic circuitry to assist in estimating the degree of correction needed to account for quenching.

After a cocktail is prepared, it is enclosed in a glass or plastic vial. Glass vials should have a low potassium content to reduce the background counts produced by naturally occurring ^{40}K, which is radioactive (it emits beta particles). To further reduce the ^{40}K background, glass vials should be very thin (and should therefore be handled carefully). Plastic vials are popular because plastic contains no potassium, and the vials therefore have a lower radioactive background than glass vials. They also have a slightly higher efficiency for ^{3}H counting. The disadvantages of plastic vials are that they are permeable to toluene, a commonly used solvent; therefore, counting rooms or laboratories in which plastic vials are used should be well ventilated. Some plastics also swell with time, which may preclude counting a sample again at a later date.

Vials containing the cocktail are placed in a lightproof enclosure containing one or more photomultiplier tubes. Quenching effects, and the fact that this counting method typically involves low-energy radiations, may produce pulses that correspond to no more than a few electrons in the photomultiplier tube. Noise (pulses arising from sources other than the radioactive sample) may also interfere with accurate and reproducible counting of the sample. Significant sources of noise include photoelectrons that are generated by heat production within the photocathode, and chemiluminescence, or additional scintillations caused by chemical reactions in the cocktail.

Sources of noise can produce extraneous photoelectrons that are included in the pulse and are difficult to discriminate against when the primary pulse (from the radioactive sample) is produced by only a few photoelectrons. The practical counting efficiency of a liquid scintillation counter is determined by its ability to distinguish between the primary pulse and the noise.

The counting interference caused by noise from the photomultiplier tube can be eliminated by using two photomultiplier tubes placed on different sides of the scintillator vial, and counting only those pulses that are observed at the same time by both tubes. Pulses arising from only one tube, which would be noise, are not counted.

Because of the efficiency and uniform geometry of liquid scintillation counting, its most common application with respect to alpha particles is for counting low-activity environmental samples. The relatively high-energy alphas have a much higher light output than the low-energy betas, and noise interference is not a problem. The energy resolution, however, is poor compared with the resolution that can be achieved using the semiconductor diode detectors discussed below.

(3) Plastic Scintillators. Plastic scintillators are solid organic solutions that are sometimes used for beta counting. They can be made much larger than organic crystal scintillators and are easily handled and shaped. A disadvantage that limits their use, however, is that they have much lower counting efficiencies than organic crystals of equal size.

2.2.3 Semiconductor Detectors

A semiconductor, or solid-state detector, is a solid crystalline material that has an electrical conductivity between that of insulators (nonconducting) and good conductors such as metals. The electrical conductivity of the semiconductor changes, however, when it is exposed to radiation, and the degree of change is related to the radiation exposure. The semiconductor detector operates on the same principle as the gas ionization detector; that is, ionizations produced within the sensitive volume of the detector cause a voltage pulse within the detector, which is then amplified and counted on a scaler system. In the semiconductor detector, a solid replaces the fill gas of the gas ionization detector, and the phenomenon of gas amplification (the production of secondary ions) does not occur. However, the voltage pulse from a gas-filled detector is smaller than the pulse from a semiconductor detector because the solid material in a semiconductor produces 10 times as many primary ion pairs as does the gas in a gas-filled chamber.

The atoms of semiconductor materials usually have four electrons in their outermost shell (i.e., four valence electrons); examples of these materials are germanium and silicon crystals. In the production of semiconductor detectors, other elements are added to the semiconductor materials. These elements are called impurities because the semiconductor crystal is no longer pure after they are added. The introduction of impurities such as lithium, aluminum, or boron, which have three valence electrons, produces a total of seven valence electrons. An atom with eight valence electrons is very stable. A material with a configuration of seven valence electrons has a space, or hole; because it wants to accept one more electron, it is called a positive or p-type material. If an impurity with five valence electrons, such as arsenic, is added to the semiconductor material, the result is nine valence electrons, or one more than the stable configuration of eight. In this case, the resulting material wants to give up its extra electron to become stable and is called a negative or n-type material, or an electron donor.

When n-type and p-type materials are combined, the extra electrons in the n-type materials combine with the holes in the p-type materials, creating electron-hole pairs and forming an electrical potential across the junction. This small potential difference is then enhanced by applying an external electric field to oppose the natural motion of the electrons and holes. This "reverse bias" is applied by connecting the positive pole of a battery to the n side and the negative pole to the p side. The depletion layer that is thus set up is the sensitive volume of the detector (see Figure 2.4). When a charged particle (alpha or beta particle) loses its energy within this depletion region, electrons are released and are attracted to the positive electrode. This movement produces a current pulse that can then be amplified and electronically measured with considerable accuracy. The diffused p-n junction detector is not useful for detecting photons because the depletion layer is only a few millimeters deep.

The germanium-lithium detector, or GeLi detector (pronounced "jelly"), and the silicon-lithium detector, or SiLi detector (pronounced "silly") are two examples of semiconductor detectors that operate on the same principle as diffused junction detectors but that have a much larger sensitive volume,

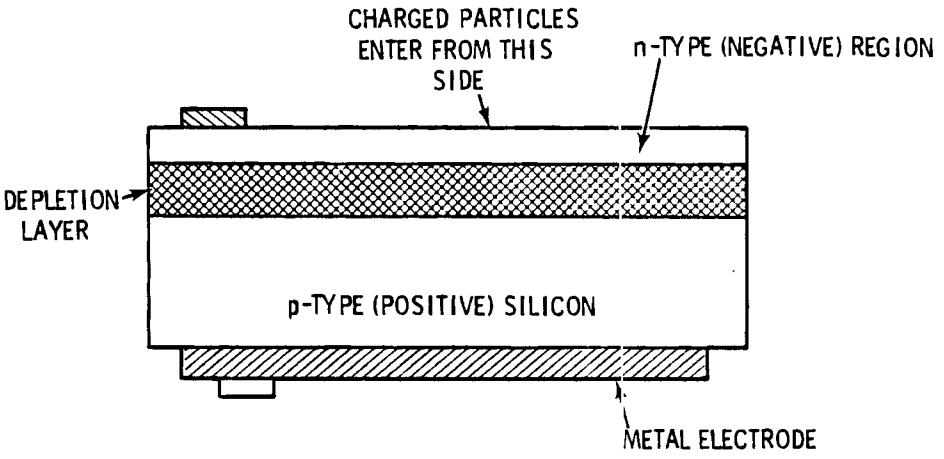


FIGURE 2.4. Diffused p-n Junction Detector

which makes them suitable for gamma counting. Lithium is drifted into a p-type germanium or silicon crystal by heating the crystal and applying a reverse bias across it. A wide layer, called the intrinsic or compensated layer, is formed where the lithium, which denotes one valence electron, exactly compensates the p-type material. This is the sensitive volume of the detector, and thicknesses of more than 1 cm can be achieved. GeLi detectors must be kept cold using liquid nitrogen (the detectors are designed to hold this coolant) because the lithium tends to "redrift" if the crystal is allowed to warm up to room temperature. SiLi detectors can be operated at room temperature but they have a relatively low counting efficiency compared with GeLi detectors because of their lower density.

Semiconductor detectors of the GeLi and SiLi type are most frequently used for gamma-ray spectroscopy. They have the ability to differentiate, with a high degree of resolution, among various energy peaks. Semiconductor detectors have a lower counting efficiency than sodium iodide crystals. However, their energy resolution is far better than that of sodium iodide detectors because of the long sequence of events that takes place in the sodium iodide detector to convert the radiation to light and then to an electrical signal. Semiconductors detectors are relatively expensive, and because of their fragile nature and design, they cannot be decontaminated.

Section 2.3 CALIBRATION OF INSTRUMENTS

The performance and accuracy of radiation detection and measurement instruments depend on the design characteristics of each instrument and on proper calibration and reliability checks made during its use. Calibration is the evaluation of an instrument's response to the type and energy of radiation it was designed to detect or measure, as well as to any other radiation that may be present and contribute to the radiation reading. Calibration also involves examination of the instrument's electrical and mechanical integrity. The AN/UDM-2 calibrator, which is intended to calibrate tactical instruments, should not be used to calibrate instruments used for radiation safety.

The extent of a radiological calibration operation at an installation depends largely on the requirements of the radiation protection organization. The funds available to a radiation protection office may limit the availability of facilities, calibration sources, and technical staff for radiological calibrations. If, for any of these reasons, an office is unable to provide a proper calibration program, the RPO should seek outside assistance from another command or from a commercial calibration service, rather than permitting the quality of the calibration services provided to be compromised.

2.3.1 Calibration Sources

The foundation of a good calibration program is the use of standard radiation sources that have well-defined properties and are traceable to the National Bureau of Standards (NBS). Such sources can be obtained in three ways:

1. They can be purchased from a vendor.
2. In certain cases, the installation's own sources (e.g., small neutron sources) can be shipped to NBS for direct calibration. Because of the time, cost, and complication in transportation, this procedure is not frequently used.
3. An intercomparison transfer standard can be obtained by sending an ionization chamber to NBS for direct calibration with their primary standard. The NBS "certifies" the calibration and accuracy of the instrument as a

"secondary standard." The chamber, which is referred to as directly traceable to NBS, is then used to calibrate radiation sources at its home facility, and the calibrated sources are used to calibrate the facility's instruments. Sources and instruments calibrated against a secondary standard are assigned an allowable error that is larger than that of the secondary standard.

The types of radioactive sources used to calibrate instruments and dosimeters vary depending upon the needs of the radiation protection program. To ensure that the proper sources are being selected, one of the following standards documents should be referred to when calibration facilities are being designed and when calibration frequencies and the types and strengths of radionuclides suitable for the instrument calibration process are being determined: American National Standards Institute (ANSI) Standards N323-1978, N42.3-1949, and N13.11-1978.

As part of a routine quality assurance program (see Chapter 15), the activity of sources should be checked periodically. Verifying the activity of a source that will be used as a radiation standard requires absolute counting methods and the use of accurate detectors with known counting efficiencies. Sources that emit alpha and beta particles can be verified by placing the source in a gas-flow proportional counter, thus providing 4π geometry for the counting. A well-type ionization chamber (in which the source is completely surrounded by the detector) is frequently used for standardizing short-lived gamma-ray sources.

2.3.2 Calibration Facilities

Radiation calibration facilities should be located where the radiation background is low, the radiation field is well known, and conditions are stable. Facilities should be constructed of a material that minimizes scatter and should be large enough to allow for good geometry when calibrating instruments that measure photons and neutrons. General criteria for facility design are discussed in Chapter 8.

2.5.3 Instrument Characteristics That Affect Calibration and Calibration Frequency

Under certain conditions, the ability of health physics instrumentation to measure radiation accurately is limited by the equipment and its operating characteristics. Some of these conditions create a relatively small error while others could, if not recognized, put the radiation protection staff and radiation workers in jeopardy. For example, as discussed earlier, a GM detector saturates and reads zero in a high-radiation field. As another example, a standard ionization chamber often produces a false reading when used around a source with a three-phase alternating current (e.g., a three-phase x-ray machine). An ionization chamber that is compensated for radio frequency must be used to avoid this problem.

The size of a source and the distance between the source and the instrument also affect measurement accuracy. If the source is not a point source and the distance between the source and the detector varies, corrections for source size and source-to-detector distance need to be developed and used. Curves illustrating these corrections are supplied by some instrument manufacturers upon request. If they are not available, they can be generated by a qualified health physicist. An effective calibration program should include the assignment of proper correction factors for each instrument type used in the radiation protection program. The correction factors should be based on the range of sizes of radiation fields and the source-to-detector configurations commonly used for each instrument type.

One of the primary factors affecting the accuracy of any measurement (either in calibrations or in field use) is the position of the source relative to the position of the sensitive volume of the detector, that is, whether the entire sensitive volume is being irradiated. It it is not, then geometry correction factors must be applied to the instrument readings. Part of the contribution to geometry errors is the difference in the radiation field during actual use and during calibration. Exposure rate instruments are usually calibrated in a radiation field of nearly uniform intensity. However, in many actual field situations, these detectors are used in nonuniform fields (i.e., close to a source) or are not entirely exposed. In either of these

actual-use situations, the response of the instrument can be low by a factor of 50. Under normal conditions, underestimation factors of 10 and above may occur.

Limitations associated with the ability of an instrument to accurately measure both high and low radiation energies are known as the energy dependence of the instrument. Energy dependence can be caused by many factors. If high-energy radiation causes photoelectrons to be emitted from the detector wall and the instrument reads them, then the total instrument reading is high. If low-energy radiation is absorbed by the detector wall, then the instrument reads low. The energy dependence of an instrument can be evaluated by exposing it to identical exposure rates from NBS-traceable sources that emit different photon energies. An instrument correction factor for a given energy can be calculated by dividing the measured exposure rate by the true exposure rate. Curves of correction factors versus radiation energy are usually available from the instrument manufacturer.

If the measurements made with an instrument vary significantly when the instrument's position is rotated through a radiation field, the instrument is considered to have angular dependence. Angular dependence may cause serious discrepancies in instrument readings, particularly if the instrument is not properly positioned in the radiation field. If angular dependence appears to be a problem for an instrument, the instrument should be calibrated at 15° increments in a full 360° plane perpendicular to the source.

During the calibration process, portable survey meters should be tested to ensure that they respond only to the type of radiation they are designed to detect. That is, alpha or neutron monitors should be verified to be insensitive to photon radiation. Similarly, photon monitors such as ionization chambers should be insensitive to other forms of penetrating radiation such as neutrons. Also, scintillation detectors should be closely checked with a high-intensity light source to verify the absence of light leaks that could produce a false count.

The frequency and extent of routine instrument calibrations are governed by many factors, including the rate at which components in each instrument age or become damaged. The ANSI standards listed earlier in this section describe

the process used in establishing calibrations frequencies. They also describe procedures for simple constancy checks to be used between calibrations.

Section 2.4 FACTORS THAT AFFECT THE SELECTION AND USE OF RADIATION-MONITORING INSTRUMENTS

Individuals who are selecting instruments for radiation monitoring should know the purpose for which the instrument will be used, the degree of accuracy needed, the type of radiation to be detected or measured, the energy of the radiation, the source form (whether solid, liquid, or gaseous), and the intensity and uniformity of the field to be measured. Knowledge of these parameters and of the limitations of various types of radiation detection and measurement devices will ensure the selection of the best instrument for each application. Each facility should have on hand a list of available radiation survey instruments, including the types of instruments available and, for each type, the number available, the radiation it detects, its sensitivity and range, the thickness of any windows, and the general use it was designed for. This listing, together with the calibration date on each instrument, can assist in the selection of the best available instrument for each situation.

Several of the factors that should be considered in the selection and use of radiation monitoring instruments are discussed briefly below.

2.4.1 Detection Versus Measurement

The purpose for which an instrument will be used and the accuracy required dictate which instrument should be selected. An instrument designed only for detection should not be used to measure radiation dose rate or exposure rate.

2.4.2 Type of Radiation

A principle factor in the selection of an instrument is the type of radiation to be detected or measured. For example, a specially designed GM counter can detect alpha, beta, and gamma radiation, but a portable alpha counter that is property calibrated should not measure gamma radiation. A standard ion chamber measures beta and gamma radiation but does not detect neutrons. A rem meter detects neutrons but does not detect external alpha particles. If an

instrument is sensitive to several types of radiation, either mechanical devices (shields or filters) or electronic discriminators can be used to distinguish between the various types of radiation.

2.4.3 Radiation Energy and Instrument Energy Dependence

The instrument selected must be capable of measuring the radiation in question. Most instruments are designed to respond to a wide energy spectrum (e.g., 150 eV to 3 MeV). However, a GM counter or an ionization chamber cannot monitor a substance such as tritium; the weak beta radiation (18.6 keV) emitted by tritium requires measurement by liquid scintillation methods or special windowless counters.

The most reliable method of determining whether an instrument operates accurately in the energy range of a specific radionuclide is to attempt to calibrate it against the radionuclide. Because each instrument will respond differently, it is useful to provide calibration curves, especially for beta calibration.

2.4.4 Nonuniform Fields

The quantification of radiation exposure rates from nonuniform fields may require special calculations and the use of correction factors. Nonuniform fields can be expected when measuring 1) dose rates at the surfaces of materials, 2) plane circular sources that are smaller than the diameter of the detection chamber, 3) surface-contaminated cylinders such as rods, pipes, and cables, and 4) radiation beams smaller than the diameter of the detection chamber. Correction factors for these special conditions may range from 1 to over 100 depending upon the condition, the type and energy of the radiation, and the particular instrument being used. Special studies and consultation with experienced health physicists may be needed.

2.4.5 Angular Dependence

If the direction from which radiation arrives at an instrument differs significantly from the directions used in the calibration field, correction may be necessary. Instrument response may be extremely directional for some instruments and radiations; for others, directional effects may be relatively

insignificant. Radiation protection personnel should be alert to the potential for directional response and should provide corrections if necessary.

2.4.6 Calibration

The selection of an instrument should be based on the instrument's demonstrated capabilities, including its ability to be calibrated. Before any instrument is placed in field service, a thorough calibration and operational check should be performed, including verifying that batteries are fully charged.

2.4.7 Unwanted Response

A portable survey instrument's response to stimuli other than the radiation it is supposed to measure constitutes what is called unwanted response. Instruments may respond to heat, light, radio frequency radiation, and mechanical shock. When used near operating equipment, particularly vehicles with generators or alternators, survey instruments may respond to induced electrical fields. In some instances, components of an instrument (other than the detector itself) may respond to radiation, causing measurement errors. This response is called extracameral sensitivity.

Section 2.5 TYPES OF RADIATION-MONITORING INSTRUMENTS

Radiation-monitoring instruments are generally classed in one of four areas, depending upon their particular application: 1) portable survey meters; 2) laboratory counting instruments; 3) air-monitoring equipment; and 4) other fixed instruments. The uses of these four classes are discussed below.

2.5.1 Portable Survey Meters

Portable survey meters are instruments small and light enough to carry from place to place. Some are used for detecting radiation and radioactive materials, and others for quantitatively measuring radiation levels. In both cases, some degree of accuracy and precision must be sacrificed to provide the light weight, small size, and ruggedness necessary for portable instruments.

For measurements of very low levels of activity, such as many measurements of environmental samples, or for measurements requiring a high level of accuracy, laboratory conditions and laboratory counting equipment should be used.

A. Portable Detection Instruments. Portable survey meters for detecting radiation or radioactive materials (e.g., GM courters) should be selected based upon the type, energy, and intensity of the radiation to be encountered. Most portable detection instruments are count rate instruments. They frequently incorporate a meter display and an aural output, using earphones or a speaker or both. For surveys of areas, equipment, or personnel, the aural output should be used if it is available because the aural circuitry of these instruments responds more rapidly to radiation increases than does the meter circuitry. Small radioactive spots or beams can be more readily detected by sound than by observing the meter movement. In addition, the aural circuitry does not fail if the device saturates and the meter indicates zero.

Even though portable survey instruments are relatively small and rugged, they must be handled and used carefully to prevent damaging them while still effectively detecting radioactivity. Most instrument detectors or probes have a very thin window or covering over the sensitive detector area or the probe. Puncturing this window may cause an implosion in some detectors (GM tubes) or light leaks that lead to erratic response in others (scintillation detectors). For this reason, most detectors have a screen or grid protector over the window. This screen helps protect the window, but it also reduces the sensitive window area.

B. Portable Measurement Instruments. Portable survey instruments for measuring exposure or exposure rate are generally small, portable ionization chambers. Like portable detection instruments, portable measurement instruments are selected based on the type, energy, and intensity of the radiation to be measured, and the degree of accuracy needed. The technical specifications of an instrument should be reviewed to determine whether it is appropriate for a particular use. In addition, the methods and radioisotopes used to calibrate the instrument, the calibration curves, and the necessary correction factors all affect the suitability of an instrument. Table 2.1 summarizes the kinds of portable survey instruments available for both detection and measurement of radiation.

TABLE 2.1. Portable Survey Instruments

Instrument	Range of Counting Rate	Radiations Detected	Typical Uses
Air proportional counter with probe	0 to 100,000 dpm over 100 cm^2	α, photons	Surfaces, hands, clothing
Gas-flow proportional counter with probe	0 to 100,000 dpm over 100 cm^2	α, photons	Surfaces, hands, clothing
Portable scintillation counter with probe	0 to 100,000 dpm over 100 cm^2	α, β, γ	Surfaces, hands, clothing
Portable count rate meter (thin-window GM counter)	0 to 1,000,000 cpm	β, γ (α-sensitive with appropriate detector probe)	Surfaces, hands clothing
Portable count rate meter (BF_3 tube)	0 to 500,000 cpm	Neutron	Area, beams

2.5.2 Laboratory Counting Instruments

Field assessments of radioactive contamination are generally qualitative rather than quantitative, and even when portable measurement instruments are used, they cannot measure levels of radioactivity as low as the levels laboratory counters can measure. To precisely quantify levels of activity, laboratory conditions and laboratory counting instruments are required. Laboratory counters may include GM tube detectors in heavily shielded chambers with scaler readouts, scintillation counters, proportional counters, semiconductor detectors, and multichannel spectrometers with computer analysis capabilities. The counter selected for a specific application depends on the type, energy, and level of radiation to be measured, and on the accuracy and precision required. Certain laboratory counting instruments (e.g., NaI crystals) can be used to determine the particular radionuclides in a sample as well as to measure the activity of each radionuclide.

Table 2.2 lists some of the available laboratory counters and their sensitivities, as documented in Report 57 of the National Council on Radiation Protection and Measurements (NCRP 1978). Most samples analyzed as part of radiation protection programs contain very small amounts of activity. The

TABLE 2.2. Laboratory Counters[a]

Instrument	Radiations Detected	Sample Sensitivity (μCi)
GM counter	β	10^{-4}
	γ	10^{-2}
Gas-flow proportional counter	β	10^{-5}
Gamma scintillation counter		
Well	γ	5×10^{-5}
Probe		10^{-2}
Liquid scintillation counter	β	10^{-5}
Alpha scintillation counter	α	5×10^{-4}
Semiconductor	α	<1 dpm
	γ	5×10^{-5}

(a) NCRP 1978.

counting instruments used should therefore be highly sensitive, and the effect of natural background radiation levels on the detectors should be kept as low as possible. Facilities used for laboratory counting should be located in areas of low background. Room or detector shielding may be required to reduce instrument background levels.

Extra precautions should be taken to assure that laboratory counters are not contaminated by the samples being counted. Because these instruments are highly sensitive to radiation, very small amounts of contamination bias their counting results. Frequent verification of background counting levels is necessary. In counters that have reusable sample holders, or planchets, the empty sample holders should be counted periodically to ensure that they have not become contaminated.

2.5.3 Air-Monitoring Equipment

Instruments used to monitor gaseous or particulate radioactivity in air should be highly sensitive because the amount of activity to be detected or measured is usually small. The type of equipment used depends upon the type, energy, and half-life of the radiation to be detected, whether it is in gaseous or particulate form, and whether sampling or monitoring is to be done.

Air sampling and air monitoring are both performed to determine the presence or amount of radioactivity in air. An air sampler either collects the air (for sampling radioactive gases) or pulls the air through a filter (for sampling radioactive particulates in air). In either case, the sample is removed for later analysis. An air monitor, on the other hand, analyzes the air in question as it is collected.

A. Air Samplers. Air sampling is performed in the following circumstances: when the probability for airborne contamination is low; as part of a long-term environmental program; where a high level of background radiation or excessive contamination prohibits air monitoring; when the consequences of airborne contamination are known not to be of immediate concern to the personnel in the area; as a check on the monitoring program; where great sensitivity for radionuclide identification is required; and where surrounding conditions (e.g., potentially explosive atmospheres) do not allow the use of monitoring equipment. The advantage of an air-sampling system is that the sample can be taken to an area of low background radiation, where it can be evaluated or held for the decay of natural radioactivity, if desired, and where various sample-processing steps can be performed and sophisticated equipment can be used to analyze the sample.

A general-purpose air-sampling system consists of a collector (filter or sorbent), collector holder, flow-measuring device, flow rate controller, and air mover. Some gas-sampling systems use evacuated flasks, cold condensate traps, or specially treated traps (e.g., activated charcoal for sampling radon gas). Most sampling systems have the advantage of being small and portable.

In some areas, small battery-operated samplers (lapel samplers) can be carried or worn by individuals to provide an integrated sample of the

contaminants in the individual's breathing zone (the air directly surrounding the face). Fixed samplers can also be selectively located to provide long-term integrated samples, which are useful in establishing the average concentration of contaminants near probable points of release and throughout the work area.

Grab samples are usually high-volume samples collected over a short time (i.e., from 2 to 20 minutes) and used for determining the level of particulate contamination in air. A portable air suction pump containing a filter paper holder is located at the point of interest, and a large volume of air (2 to 100 m^3) is drawn through the filter. The filter is then removed to a counting room or laboratory for rapid analysis. Low-volume air samplers are used in environmental programs because they can be operated continuously for weeks or months at a time. When analyzed, the filters from these samplers indicate the total release from a specific site over a given period.

B. Air Monitors. Air monitoring is performed when the sampling results are needed immediately; when a real-time monitor is required to indicate the need for immediate evacuation of a work area; to provide a continuous reading for trend analysis; to monitor releases to the environment (as in stack monitoring); and to measure immersion doses from gaseous releases.

An air-monitoring system is basically the same as an air-sampling system except that an appropriate counter (e.g., a proportional counter) or other evaluation instrument is placed near the collecting medium (filter paper or sample chamber). Air monitors are often equipped with strip-chart recorders, air activity meters (which indicate, for example, cpm per liter of air), check sources, and visual and audible alarms. The advantage of an air-monitoring system is its continuous and immediate indication of activity levels.

Most air monitors cannot detect low levels of radioactivity; therefore, these monitors are most useful where the potential for large radioactive releases is highest. For example, an alpha air monitor is relatively ineffective for measuring airborne depleted uranium (DU). By the time an alpha monitor detected DU and sounded an alarm, the airborne activity would be several times above acceptable limits.

Some airborne activity, such as low-energy beta particles from tritium, can be measured using a Kanne chamber, an ionization chamber through which the air flows. The beta particles are drawn into the chamber, where the ionization must occur if it is to be detected. However, these chambers are also sensitive to higher-energy background radiation, and some compensation for background is normally required.

C. Principles of Operation. Air sampling and monitoring involve collecting a sample of air or a material removed from the air and determining by analysis what the contaminant is (if that is not already known) and the quantity of it. Accurate determination of the activity in a sample requires accurate measurement of the volume of air sampled. For gaseous samples, this may be as simple as knowing the volume of the chamber in the sampler used. However, a system for sampling particulates requires accurate measurement of 1) the rate at which air flows through a filter medium and 2) the time over which the sample is taken. The system must have an air mover capable of moving the air at the rate desired, a method of ensuring that the air flow is constant for the sampling period, and calibration of the air sampler.

Many variables must be considered in establishing a quantitative air sampler. The type of filter paper or sorbent medium should be selected to effectively remove from the air the contaminants of interest. The collection efficiency of the medium should be established, taking into account the size of the particles collected and the air velocity during collection. Isokinetic sampling of ducts and effluent stacks should be used. This means that the opening of the sampling device should be set perpendicular to the direction of air flow, and the sample flow rate should be adjusted so that the linear air speed into the sampler is the same as that of the approaching air stream. Anisokinetic conditions may cause an over- or underestimation of particulate air concentrations in the air stream. In addition, the representativeness of the sample at the collecting point may be affected by materials becoming deposited on the sampling lines or passages, a condition called plateout. Attention must be given to limiting the length of a sample line, the degree of curvature of bends in the line, and the temperature gradients between the air being sampled and the line.

Instrumentation used to measure the activity of the collected sample is selected based on several factors: whether the instrument is to be used as part of a continuous monitor or whether it is in a counting room or laboratory; the type and energy of the radiation being detected; and the sensitivity required. Geiger-Mueller counters, gas proportional counters, scintillation counters, or semiconductor counters can be used to measure the activity of air samples.

2.5.4 Other Fixed Instruments

In addition to the radiation detection and measurement instruments previously discussed, special-purpose instruments can be used. These instruments include remote area monitors and continuous air monitors.

A. Remote Area Monitors. Remote area monitors (RAMs) are usually GM detectors or ionization chambers used to monitor direct exposures to photon radiation. These monitors are usually permanently positioned and have visual or audible alarms or both. They are often connected to other RAMs in a network, with the results displayed in a central control room. These monitors usually have a variable alarm setting so that the alert level can be altered.

In addition to the alarm function, RAMs may incorporate a continuous recorder so that a historical record of radiation levels is provided and radiological conditions and trends can be followed and evaluated.

B. Continuous Air Monitors. Continuous air monitors (CAMs) are similar to remote area monitors in function, but they always monitor the radioactivity concentrations in air continuously. This type of air monitor can be fixed in place, with sample lines to the instrument from the area being monitored, or it can be semiportable (usually a relatively heavy cart on wheels) and can be moved to the area to be monitored. Depending upon the type of radiation to be measured and whether it is in gaseous or particulate form, CAMS may use GM, gas proportional, semiconductor, or ionization chamber detectors. The complete CAM unit includes an air mover, air flow controls, the appropriate electronics for the detector being used, an alarm, and usually a recorder. Those fixed in place may also be wired for a meter readout, a strip chart recording, and an alarm at some remote or central location.

The factors that affect other air-sampling and air-monitoring systems also affect CAMs. In addition, CAM units can be affected by changes in ambient radiation levels, the fluctuations of unregulated power, and contamination from outside the area being sampled.

To avoid a long-term buildup of radioactivity and dust on filter media, fixed CAMs require frequent filter changes. Other CAM units use a moving filter tape. An advantage of the moving-filter CAM is the capability of providing a delayed counting sequence to allow for the decay of natural background radioactivity. Instruments of this type can be provided with duplicate detectors, one instantaneous and one delayed, and electronic circuitry to allow background compensation and alarm functions for both instantaneous releases and long-term buildups of radioactivity.

Section 2.6. PERSONNEL DOSIMETERS

A radiation dosimeter, loosely defined, is any instrument or system capable of measuring radiation dose. Dosimeters are typically used to provide a quantitative estimation of the radiation dose actually received by personnel. Their response should be reproducible, precise, and accurate, and the instruments should be able to measure all ionizing radiations encountered by personnel. They should be simple and convenient to use, small, easy to handle, and low in cost. Because personnel dosimeters record only the dose they have received, it is extremely important that personnel be trained in their proper use. One type of dosimeter, the pocket ionization chamber or pencil dosimeter, was already discussed in Section 2.2.1. Three other types--photographic film, nuclear track emulsions, and thermoluminescence dosimeters--are discussed below.

2.6.1 Photographic Film

Photographic film is measurably darkened by radiation, and can therefore provide a useful estimation of personnel exposure. The response of photographic film depends on the type, energy, and amount of the radiation reaching the film.

A. Principles of Operation. The sensitivity of film is defined as the amount of darkening produced by a specified radiation exposure. Photographic films, or emulsions, consist of a layer of tiny silver halide crystals embedded in a gelatin matrix. The emulsion is spread across a thin sheet of plastic or glass plates. The thickness of the emulsion can range from 10 to 2000 μm, depending on the sensitivity desired. The thicker the emulsion, the greater the sensitivity of the film.

When ionizing radiation travels through photographic emulsions, the radiation imparts a small amount of energy to the silver halide crystals, causing some of the silver ions to be reduced to free atomic silver. These silver atoms form traps capable of capturing electrons, which can in turn reduce more silver ions and create a microscopic aggregate of silver atoms. These silver aggregates are frequently referred to as latent image centers. Chemical treatment of the film causes the latent image centers to be reduced to metallic silver, which appears to the eye as a blackening of the film. The degree or density of darkening can then be related to radiation exposure.

B. Dosimeter Design. Photographic films are incorporated into the so-called film badge. The modern film badge is designed so that radiation can reach the film either directly through an open window, or through filters. The filters are disks made of metals, such as lead, tin, copper, cadmium, silver, or aluminum, and are used to distinguish between different types and energies of radiation. For example, thin filters of a low-atomic-number (low-Z) material, such as aluminum, can be used to distinguish between gamma rays and high-energy beta particles. Other metallic filters can help identify the contribution of different components of the gamma-ray spectrum. Most film wrappers stop beta particles with an energy less than about 150 keV. Thus, film cannot be used to monitor radiation exposures from low-energy beta emitters such as ^{3}H and ^{14}C.

C. Effects of Environment. Photographic film degrades with age. Under normal conditions, dosimeter films usually last for several months before they begin to deteriorate. However, the latent image centers and the overall response of the film can be adversely affected by environmental conditions. The latent image fades if the film is subjected to high temperatures, high humidity, or oxygen. Of all these influences, relative humidity is the

dominant factor. Film packets should not be used or handled by unqualified personnel. Films should be kept in their lightproof packages to reduce the possibility of light leaking in, which could ruin the film.

D. Processing Techniques. The process used for developing film dosimeters is basically the same as that used for developing medical x-ray films. Specifically, a film is placed first in a developer solution and then in a fixer, which stops the development process by dissolving the unused silver halide crystals. How long the film is left in the developer solution, the amount of agitation of the solution, and the temperature and age of the solution all affect the first step of the process. How long the film is left in the fixer affects the quality and permanence of the image on the film. When the film is removed from the fixer (after approximately 10 minutes), it is washed and then dried at room temperature.

E. Interpretation and Calibration. Once the film has been processed, it is read and interpreted. To reduce the probability of error in the reading of the film, unexposed control films are processed along with the exposed films. Unexposed films produce a density or darkening during processing known as the base fog. By processing control (unexposed) dosimeters along with the exposed dosimeters, it is possible to subtract the degree of darkening of the base fog from the degree of darkening on the exposed dosimeters.

The processed film is analyzed using a densitometer, a device that measures the degree of film darkening. Interpretation of the densitometer reading is then related to exposure, depending on the density value under each of the filters in the badge. Doses should be interpreted only by personnel who are highly skilled in evaluating photographic film. Even with properly designed filters and film badge holders, the accuracy of photographic film is limited because its response is dependent on the radiation energy and the inherent variability in films. In mixed radiation fields (fields that include both high- and low-energy radiation), low energies can result in errors of ±50% to ±200%. However, with properly designed film badges and properly controlled usage, photographic films can achieve an accuracy of ±25% in most personnel dosimetry situations.

Photographic film dosimeters are not absolute devices and therefore must be calibrated against a known source in order to relate the film density to the exposure delivered. The calibration of dosimeters should be performed under carefully controlled laboratory situations using sources traceable to NBS.

2.6.2 Nuclear Track Emulsions

Standard photographic film badges are not designed to respond to neutrons. However, nuclear track emulsion (NTA) film, which is thicker than standard photographic film, can be used to monitor for neutrons. The neutrons reaching the NTA film interact in a variety of ways with the atoms in the emulsion, charged particles are produced, and the charged particles in turn interact with the silver atoms of the NTA film to form tracks that are visible after the film is developed. The tracks can be counted and related to neutron dose.

Nuclear track emulsions are even more sensitive to latent image fading than are the standard films. Therefore, the wearing interval for NTA film dosimeters normally does not exceed 2 months, and 2 weeks is the preferred wearing time in a high-humidity climate. Fading can be reduced and the wearing time increased if the NTA film is sealed into a moisture-proof package in a nitrogen atmosphere.

2.6.3 Thermoluminescence Dosimeters

Some crystals emit light when they are heated after exposure to ionizing radiation; this process, known as thermoluminescence, is similar to the scintillation process described earlier and is the basis for another type of personnel dosimeter.

A. Principles of Operation. The crystals most commonly used in thermoluminescence dosimeters today include lithium fluoride (LiF), calcium fluoride (CaF_2), calcium sulfate ($CaSO_4$), and lithium borate ($Li_2B_4O_7$). When one of these crystals is exposed to ionizing radiation, many of the free electrons within the crystal become excited and are caught in imperfections of the crystal, or traps. The exposed crystal can be stored at room temperature for long periods without a significant number of the electrons escaping from the

traps. However, when the crystal is heated to higher temperatures, the trapped electrons escape and lose their excess energy by the emission of visible light (thermoluminescence). Because the amount of light released from a heated crystal is proportional to the energy or radiation dose absorbed within the crystal, the phenomenon of thermoluminescence can be used in radiation dosimetry. A dosimeter that uses this phenomenon is called a thermoluminescence dosimeter (TLD).

A TLD reader, which has a controlled heating element, is used to determine how much light is emitted during the heating of a dosimeter crystal. The light intensity is plotted as a function of temperature, and the resulting graph is called a glow curve. The glow curve normally has several peaks at various temperatures. The area under any peak can be used as a measure of the dose received by the TLD.

When a TLD has been irradiated and read on a TLD reader, it can be annealed and reused. Annealing is a slow heating process that completely empties the traps and restores the crystal to its original state. After the crystal has been allowed to cool, it is ready to be reused.

B. Advantages and Limitations. The TLD has a wide dose-response range (1 mrad to 10^5 rad) and a very low energy dependence. The most popular TLD material, LiF, has an effective atomic number very close to that of human muscle tissue. Thus, it is considered by most users to respond much as tissue would and is frequently considered "tissue equivalent."

Other advantages of TLDs are that they are very small, quite rugged, and essentially unaffected by environmental variables. Because TLDs show very limited fading (unlike film dosimeters), the wearing interval for the TLD can be a year or longer. The advantage of the longer wearing period is a reduction in the error produced by numerous processings throughout the year. The reported accuracy of most TLDs under controlled laboratory conditions is ±1%. An accuracy of ±10% is fairly easily achieved in the field.

Thermoluminescence dosimeters are essentially unaffected by their orientation in the radiation field and by the rate of exposure. However, the badge

or device that is designed to hold the thermoluminescent material may adversely affect the accuracy of the dosimeter. Therefore, proper badge design is essential in the correct use of TLDs.

A major limitation of TLDs is that, after they have been processed, their exposure information is erased; film, on the other hand, retains the information as a permanent record.

C. Interpretation and Calibration. Interpreting the results of a glow curve produced from an irradiated TLD requires establishing a relationship between the glow curve and a known exposure level. The best procedure is to obtain a large batch of dosimeters with well-matched responses and to run a calibration curve over the exposure range of interest, using a known radiation field.

The use of properly calibrated dosimeters is critical to a good health physics program. An installation that has a small radiation protection staff should procure the services provided by the Army or a commercial calibration company. Calibration companies should maintain their traceability to NBS through a periodic direct intercomparison.

D. Practical Applications. Thermoluminescence dosimeters can be used in any situation where film dosimeters are currently being used. They are preferred to film for extremity dosimeters (e.g., ring and wrist badges), for personnel monitoring where radiation energies are below about 100 keV, and for environmental monitoring. However, TLDs do not provide a permanent record of exposure, as film dosimeters do.

Unlike film dosimeters, TLDs can also be used to measure the neutron radiation to which an individual is exposed. Thermoluminescent materials are more sensitive to thermal (slow) neutrons than to fast neutrons. Thermal neutrons interact with a TLD as they pass through it to the wearer. Some thermal neutrons may be reflected back to the TLD from the irradiated individual and may interact with the dosimeter then also. Fast neutrons, on the other hand, do not interact with the TLD as they pass through it. These fast neutrons interact with the hydrogen in the wearer's tissues, where they lose their energy (become thermal). Many are then reflected back toward and interact

with the dosimeter. The reflected thermal neutrons are called albedo neutrons. Correct interpretation of albedo dosimetry requires that the radiation source, the dosimeter, and the irradiated individual be in line and that the original energy of the neutrons be known. The neutron energy or a description of the radiation source should be given to the dosimetry service interpreting the response.

Section 2.7 STATISTICS AND ERROR DETERMINATION

The spontaneous emission of radiation by nuclear processes occurs randomly in time, and all measurement and detection instruments must respond to these statistically random events. This means that the interpretation of instrument response must take into account the random nature of radioactive decay. We tend to assume that a measurement is an absolute indication of the activity of the source. However, this is usually not the case. It is more likely that only a fraction of the radiation can be detected. This error must be corrected using statistics and geometry correction factors.

2.7.1 Systematic and Random Errors of Measurement

The errors associated with radiation measurements can be divided into two types: systematic and random. Systematic errors are created in the measurement process or in the interpretation of measurement data. They are frequently caused by faults within the electronic systems of instruments. For example, low batteries or faulty electronic components could bias measurements, and the results would be considered to contain a systematic error. The primary source of random errors is radioactive decay.

2.7.2 Basic Statistical Distributions for Radioactive Decay

If a long-lived radionuclide of low activity was counted many times, and if a plot was made showing the number of times a given count rate occurred versus the count rate, the plot would be similar to the one shown in Figure 2.5. This curve is called a normal, or Gaussian, distribution and represents the distribution of count rate values obtained in successive counts.

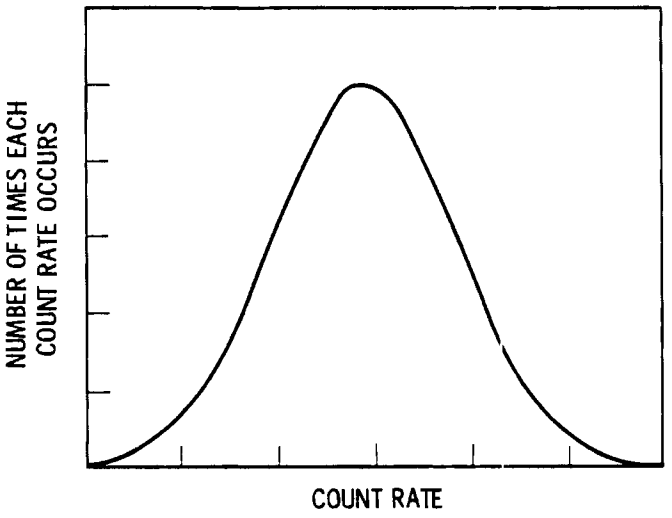


FIGURE 2.5. Frequency of Occurrence of Count Rates for a Long-Lived Sample

The normal distribution curve can be described mathematically by calculating the mean and the standard deviation of all the count rates used to prepare the curve. The mean, or the arithmetic average of the count rates, describes where on the curve the greatest number of counts occurs. It is calculated by summing all of the count rates and dividing by the number of counts taken. Written in mathematical terms, the equation appears as follows:

$$\bar{n} = \frac{1}{N} \sum_{i}^{N} n_i \qquad (2.3)$$

where $\bar{n}$ = the mean of the count rates

N = the number of times the sample was counted

n_i = the value of the ith count rate

$\sum_{i}^{N} n_i$ = the sum of all the count rates.

The individual measurements taken in any radiation survey are distributed about this sample mean.

The standard deviation (σ), a measure of variability, describes the width of the curve and is a useful indication of how extensively the count rates

vary from the average value. The square of the standard deviation is called the variance and can be approximated using the expression

$$\sigma^2 = \frac{1}{N - 1} \sum_{i}^{N} (\bar{n} - n_i)^2 \qquad (2.4)$$

where σ^2 = the variance

σ = the standard deviation

N = the number of times the sample was counted

n = the mean of the count rates

n_i = the value of the ith count rate

$\sum_{i}^{N} (\bar{n} - n_i)^2$ = the sum of all the squared deviations from the mean.

When only a few measurements have been taken (fewer than 20), a best estimate of the standard deviation can be derived as follows: $\sigma = \sqrt{n}$. When more than 20 measurements have been taken, the previous method for calculating the variance and the standard deviation should be used.

Figure 2.6 shows a plot of the normal distribution curve with several features indicated. In a normal distribution, 68.3% of all counts are within

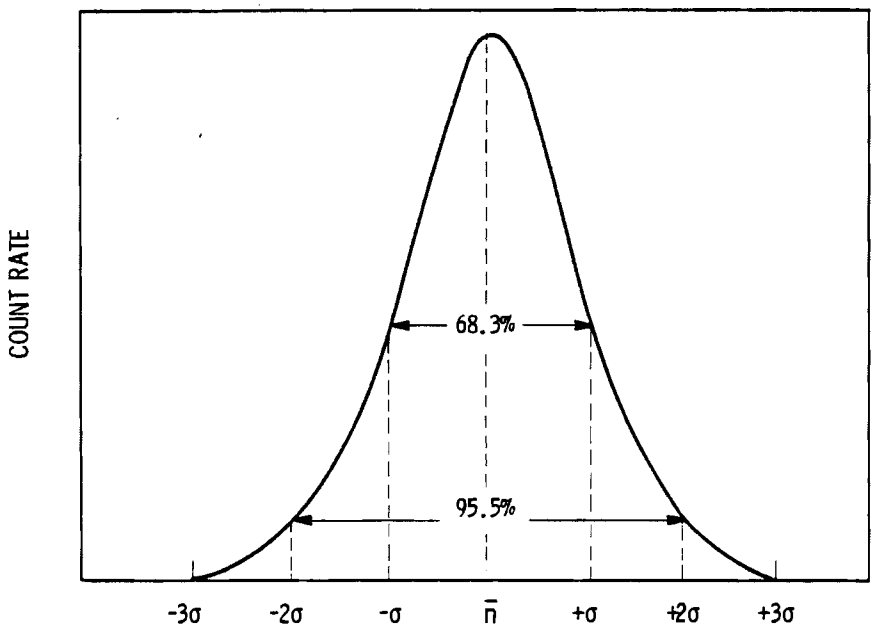


FIGURE 2.6. Normal Distribution Function Showing Standard Deviations and Mean

±1 standard deviation of the mean value, 95.5% within ±2 standard deviations of the mean, and 99.7% within ±3 standard deviations of the mean. For example, if a sample is counted 100 times, the mean value obtained is 1000 cpm, and the standard deviation is 100, then we can say, with a 68.3% chance of being correct, that the mean count rate is between 900 and 1,100. Thus, the specification of activity is a "probabilistic event"; that is, we specify with a certain statistical accuracy that the mean activity lies within a range of values.

For statistical purposes, when the results of a series of measurements are recorded, both the mean and the standard deviation should be specified.

Section 2.8 RECORDS

Records are needed to verify the availability and use of appropriate radiation detection and measure instruments, the adequacy of their calibration and maintenance, the proper interpretation and use of the resulting data, and compliance with regulatory requirements. A complete discussion of instrument recordkeeping procedures is presented in Chapter 13.

REFERENCES

American National Standards Institute (ANSI). 1949. Standard Test Procedure for Geiger-Mueller Counters. ANSI N42.3, New York.

American National Standards Institute (ANSI). 1978. Criteria for Testing Personnel Dosimetry Performance. ANSI N13.11-1978, New York.

American National Standards Institute (ANSI). 1978. Radiation Protection Instrumentation Test and Calibration. ANSI N323-1978, New York.

National Council on Radiation Protection and Measurements (NCRP). 1978. A Handbook of Radioactivity Measurements Procedures. NCRP 57, Washington, D.C.

CHAPTER 3. RADIATION PROTECTION PROGRAM

TABLES

CHAPTER 3. RADIATION PROTECTION PROGRAM

The objectives of a radiation protection program are to reduce exposures to a level as low as is reasonably achievable within the occupational dose-equivalent limits set by the federal government and the Department of the Army (DA) and to minimize the potential for accidental exposures. The components of an effective radiation protection program are common to all installations where radioactive materials are used or stored. However, the magnitude and complexity of the program may vary from one installation to another. This chapter describes briefly the principles and practices that should be considered in the establishment of a radiation protection program. These practices are covered in greater detail in later chapters of this manual.

Section 3.1 REGULATIONS

A variety of government branches and international agencies have formulated regulations governing the procurement, use, storage, transportation, and disposal of radioactive materials and sources. The National Council on Radiation Protection and Measurements (NCRP) and the International Commission on Radiological Protection (ICRP), whose members are professionals in health physics or related fields of research, provide recommendations that serve as the basis for most Army and other government agency requirements. Knowledge of and compliance with all applicable regulations are essential factors in the administration of every radiation protection program. Agencies that may have jurisdiction over specific radiological situations are discussed briefly below, and the applicability of their regulations is summarized in Table 3.1.

3.1.1 Department of the Army

All Army installations that produce, procure, receive, store, use, ship, or dispose of radioactive materials or sources are required to have a radiation protection program. Specific requirements for ionizing radiation protection programs can be found in AR 385-11, AR 40-14, DARCOM-R 385-25, and AR 700-64.

TABLE 3.1. Regulations Applicable to Army Activities

Activity	Applicable Regulations
Day-to-day operations	10 CFR 20, 10 CFR 19 AR 40-14, AR 700-64, AR 385-11 DARCOM-R 385-25
Use of radiation-producing machines (such as x-ray machines)	21 CFR 1000-1050
Transport of radioactive materials	49 CFR
Shipment through mails	39 CFR
International shipments	Inter-Governmental Maritime Consultative Organization and International Air Transport Association

3.1.2 Federal

The U.S. Nuclear Regulatory Commission (NRC) regulates the production, distribution, use, and disposal of source, byproduct, and special nuclear materials. The use of radioactive materials and radiation sources within the work environment not governed by the NRC is regulated by the U.S. Occupational Safety and Health Administration (OSHA). The requirements of NRC are described in Title 10 of the U.S. Code of Federal Regulations, Parts 19 and 20 (10 CFR 19 and 20). Army regulations require that civilian and military personnel within the United States and overseas be provided radiation protection that is at least equal to that required by 10 CFR 19 and 20.

The U.S. Department of Health and Human Services (HHS) conducts a radiation control program for electronic products. The program includes the development of performance standards to protect the public health from ionizing and nonionizing radiation in electronic products. This department also regulates and sets standards for the use of radioactive materials and radiation sources in foods, drugs, cosmetics, and medical devices, as set forth in 21 CFR.

The U.S. Department of Transportation (DOT) regulates the packaging and transportation of radioactive materials shipped in interstate commerce by air, rail, highway, and water. The U.S. Postal Service regulates shipment via the

U.S. mail. The regulations of their agencies are presented in Titles 49 and 39 of the Code of Federal Regulations, respectively.

The U.S. Environmental Protection Agency (EPA) provides federal guidance on radiation protection. The EPA also develops standards governing the release of radioactive materials and radiation sources to the environment (40 CFR).

3.1.3 International

An agency of the United Nations, the International Atomic Energy Agency (IAEA), provides overall safety guidance for the international shipment of radioactive materials. The Inter-Governmental Maritime Consultative Organization (IMCO) and the International Air Transport Association (IATA) provide regulations for the international shipment of radioactive materials. The specific application and enforcement of the regulations is the responsibility of each nation through which material is transported. Normally, a shipment that complies with the regulations of the nation of origin complies by agreement with the regulations of the nations through which the shipment is routed.

Section 3.2 RADIATION PROTECTION STANDARDS

Dose-equivalent limits for controlling occupational and nonoccupational exposure to ionizing radiation and radioactive materials have been established by DA (AR 40-14). These limits are based on the recommendations of NCRP and ICRP. Both organizations emphasize that dose-equivalent limits are upper limits for planned exposures and that every effort must be made to keep exposures below these limits and to avoid unnecessary radiation exposure. This principle is strongly emphasized in federal regulations as the As Low As is Reasonably Achievable (ALARA) philosophy.

3.2.1 Radiation Exposure Standards

Standards established by the Army fall into several categories: occupational exposures, occupational exposures to women, occasional exposures, exposure of minors, emergency exposures, nonoccupational exposures, and alternate exposure standards. These categories are described below.

A. Occupational Exposure. Occupational raciation exposure standards are presented in Table 3.2. Occupational exposure tc ionizing radiation is that resulting from military or civilian duties that directly support the use of radioactive materials or equipment capable of producing ionizing radiation. Occupational exposure does not include exposure to naturally occurring ionizing radiation or exposure received as a result of medical or dental diagnosis or treatment. An occupationally exposed individual, or radiation worker, is one whose work is performed in a radiation area or a controlled area (see Chapter 8) and who might be exposed to more than 5% of the limits given in Table 3.2.

B. Occupational Exposure to Women. Special radiation exposure controls are established for the protection of unborn children. The NCRP recommends that during the entire gestation period, the maximum dose equivalent to the

TABLE 3.2. Radiation Protection Standards[a]

Organ	Occupational Dose-Equivalent Limit, rem	
	Calendar Quarter	Calendar Year
a. Whole body, head and trunk, active blood-forming organs, gonads, lens of the eye	1.25	5
b. Skin of the whole body (other than hands, wrists, feet, or ankles) forearms, cornea of the eye, bone	7.50	30
c. Hands and wrists, or feet and ankles	18.75	75
d. Forearms	10	30
e. Thyroid, other organs, tissues, and organ system	5	15
Individuals under the age 18, and occasionally exposed individuals	10% of the values listed above	
Individuals between ages 18 and 19 (whole-body limit)	1.25	

(a) AR 40-14.

embryo-fetus from the occupational exposure of the expectant mother should not exceed 0.5 rem (NCRP Report Nos. 39 (1971) and 53 (1977)). A woman staff member is responsible for advising her employer that she is pregnant. Special consideration, such as a change in work assignment, may be necessary to ensure that her occupational exposure does not exceed recommended limits and is kept as low as is reasonably achievable. The installation commander and the Radiation Protection Officer (RPO) should determine appropriate actions and policies.

C. Occasional Exposure. An occasionally exposed individual is one whose duties do not normally involve exposure to ionizing radiation or radioactive material, but who may have a reason to enter a restricted area on a nonroutine basis. Examples are repair personnel and messengers. When such individuals enter a restricted area, they shall not be exposed to a whole-body dose equivalent of more than 1) 2 mrem in any 1 hour, 2) 100 mrem in any 7 consecutive days, 3) 500 mrem in any 1 calendar year, or 4) 5% of the values for other areas of the body detailed in Table 3.2.

D. Exposure of Minors. A minor is any person under 18 years of age. For a minor, the accumulated dose equivalent of radiation shall not exceed 10% of any of the values listed in Table 3.2. Persons over the age of 18, but who have not reached their 19th birthday, may be occupationally exposed to ionizing radiation if they do not receive a dose equivalent of more than 1.25 rem to the whole body in any calendar quarter.

E. Emergency Exposure. Radiation exposure standards in emergency situations vary according to the severity of the emergency. When entry into a hazardous area is necessary to search for and remove seriously injured persons, or to prevent conditions that may injure a number of people, the accumulated whole-body dose of each individual entering the area should not exceed 100 rad, and the accumulated dose to the hands and forearms should not exceed 300 rad. In a less severe situation, when it is desirable to enter a hazardous area to protect property, minimize the release of effluent, or control fires, the accumulated whole-body dose of each individual entering the area should not exceed 25 rad, and the dose to the hands and forearms should not exceed 100 rad. Individuals who incur such radiation exposures during an emergency

should not be allowed to do so more than once in a lifetime. The record of such exposure becomes part of the person's health record or civilian employee medical file.

F. Nonoccupational Exposure. Sources of ionizing radiation must be used in such a way that 1) the accumulated dose equivalent to the whole body for an individual person in the general public does not exceed 0.5 rem in any 1 calendar year, and 2) the average accumulated dose equivalent for a suitable sample of the exposed population or for the whole exposed population does not exceed 0.170 rem/year from all sources of radiation (excluding medical and natural background radiation).

G. Alternate Exposure Standards. Radiation exposures standards that are less restrictive than those described above may be used in special circumstances, but only when approved by the Surgeon General of the United States or the director of the Defense Logistics Agency, as appropriate. Proposals for alternate radiation exposure standards must contain a complete justification and must specify the procedures by which the standards will be implemented. Less restrictive standards will not be considered for 1) persons under 19 years of age, 2) women known to be pregnant, 3) occasionally exposed persons, and 4) nonoccupational exposure of the general public.

3.2.2 Administrative Limits and Action Levels

Administrative limits and action levels are frequently set to help maintain occupational exposures within established limits. Administrative limits are radiation exposure limits established by the administrator of a radiation protection program, for example, 80% or less of the occupational exposure standard. An administrative limit is basically a control point: as an individual's exposure approaches this level, the individual is carefully monitored so that the exposure does not exceed the limit unless specific management approval is obtained. Thus, individual exposures are kept lower, and the possibility of exceeding permissible exposure limits is reduced.

Action levels are dose-equivalent limits that, when reached or exceeded by an individual, require formal investigation into the cause of exposure. The RPO should establish investigative procedures. An investigation should lead to

the identification of portions of the radiation protection program that need to be improved. Action levels, also called investigative levels, are established by radiation protection management.

3.2.3 The ALARA Philosophy

Even though current occupational exposure limits keep the risk of injury to personnel very low, it is prudent to avoid unneccessary exposure to radiation. The operating philosophy of every radiation protection program should be to reduce occupational exposures as far below specified limits as is reasonably achievable. This philosophy, emphasized in federal regulations and referred to as ALARA (As Low As is Reasonably Achievable), means that each work procedure that will result in a radiation dose should be subject to scrutiny and that methods to reduce the dose should be identified. The methods that involve the least cost and result in the greatest reduction of dose should be considered and implemented wherever possible. References in the bibliography discuss ALARA and ALARA programs in greater detail than is possible here.

It is not desirable to maintain the dose equivalent of a radiation worker at a small fraction of the applicable limit if this practice requires that a larger number of people be exposed. Therefore, in addition to maintaining occupational exposure to individuals as far below limits as is reasonably achievable, the goal of ALARA is to keep the sum of the doses received by all exposed individuals (radiation workers, other personnel, and the general public) at the lowest practicable level. The sum of the dose equivalents received by all exposed individuals is called the collective dose equivalent.

Section 3.3 ELEMENTS OF A RADIATION PROTECTION PROGRAM

An effective radiation protection program includes licensing, an ALARA program, surveillance and monitoring programs, proper design of facilities in which radiation sources are used, control of radioactive materials and waste disposal, emergency planning, adequate training of personnel, the maintaining of reliable and complete records, and a quality assurance program.

3.3.1 Licenses, Authorizations, and Permits

Whenever radioactive materials or radiation sources are produced, procured, used, stored, transported, or disposed of at DA facilities, an NRC license and/or DA approval is required. Procedures for obtaining the necessary documents are contained in AR 385-11. Non-Army agencies, including civilian contractors, are required to obtain a DA radiation permit to possess, use, store, or dispose of radiation sources on an Army installation.

3.3.2 ALARA Program

The establishment and management of all radiation protection programs within Army facilities should be guided by the ALARA philosophy. Each radiation protection program should therefore include a formal ALARA program. An effective ALARA program requires management commitment and the assignment of ALARA responsibility to an individual or committee, as discussed below. Procedures for maintaining exposures ALARA are described throughout this manual. Particular attention should be directed to Chapters 5 and 6, which described the control of internal and external exposure.

A. Management Commitment. Management commitment to the safe and correct use of radiation and radioactive materials is probably the single most important characteristic of a good radiation protection program. Upper management, specifically the base commander, sets the tone for the safety program. The commander must indicate by word and action that safety is important. Simply displaying safety slogans and posters, holding safety contests, and establishing safety committees have little effect unless individual staff members believe that safety is important to their supervisors.

The commitment made by management to minimize exposures should result in clearly defined responsibilities for radiation protection and an environment in which the radiation protection staff can do its job properly. This commitment should be made evident in the following areas:

(1) Personnel Awareness of Management Commitment. The ALARA principle should appear in policy statements, instructions to personnel, and similar documents. Staff members should be familiar enough with this commitment to explain what management policy is, what is meant by keeping exposure to

radiation "as low as is reasonably achievable," why it is recommended, and how they have been advised to implement it on their jobs. They must understand the importance of the philosophy.

(2) Radiation Protection Personnel. Management should ensure that there is a well-supervised radiation protection staff with well-defined responsibilities. The RPO should be qualified to handle any potential problems at the installation.

(3) Training. Management should ensure that personnel receive sufficient training. Section 19.12 of 10 CFR 19 requires that personnel be instructed in radiation protection. They should understand how radiation protection relates to their jobs and should be tested on this understanding at least once each year. Radiation workers should have opportunities to discuss radiation safety with the radiation protection staff whenever the need arises. The training program in radiation protection should be reviewed by management at least once every two years.

(4) Facility Modifications. Modifications in operating and maintenance procedures and in plant equipment and facilities should be made if they will substantially reduce exposures at a reasonable cost. Management should encourage the staff to suggest improvements and modifications and should implement them where practicable.

(5) Audit Programs. A formal audit should be conducted periodically to determine how exposures might be reduced. The audit should include reviews of operating procedures and exposure records, inspections, and consultations with the radiation protection staff.

B. Assignment of ALARA Responsibility and Authority. The base commander should formally assign ALARA responsibility to an individual such as the RPO or to a group of individuals such as the Ionizing Radiation Control Committee (IRCC). The RPO should have sufficient authority to prevent unsafe practices and to communicate promptly with an appropriate level of management about halting unsafe operations. This authority should be specified in written policy statements. The members of the IRCC are chosen for their knowledge of radiation safety principles, engineering, and design, knowledge that is useful in evaluating the safety of projects involving radioactive materials.

Operating procedures related to radiation safety should be reviewed and approved by radiation protection personnel. The RPO and/or the IRCC should be responsible for conducting surveillance programs and investigations to ensure that occupational exposures are as far below the specified limits as is reasonably achievable. All of these individuals should constantly be seeking new and better ways to perform all radiation jobs with less exposure. There are several aspects of this responsibility.

(1) Monitoring of Exposures. The RPO and the radiation protection staff should know the origins of radiation exposures by location, operation, and job category and should be aware of trends in exposures. They should be able to describe which locations, operations, and jobs are associated with the highest exposures and why exposures are increasing or decreasing. Where standing operating procedures are used, exposures received should be recorded on the written procedures.

(2) Investigation of Unusual Exposures. When unusual exposures have occurred, the radiation protection staff should direct and participate in an investigation of the circumstances to determine the causes and take steps to reduce the likelihood of similar future occurrences. For each such occurrence, the RPO should be able to demonstrate that an investigation was carried out, that conclusions were reached as a result of the investigation, and that appropriate corrective actions were taken.

(3) Review of Operating Procedures. The RPO and the radiation protection staff should periodically review operating procedures that may affect radiation safety. They should survey plant operations to identify situations in which exposures can be reduced, and should implement any changes that are needed. The RPO should repeatedly emphasize that work performance that results in personnel meeting dose-equivalent limits is not acceptable when it is practical to reduce the dose to a lower level. Procedures should be established for receiving and evaluating staff members' suggestions relating to radiation protection and dose reduction, and the staff should be aware of these procedures.

(4) Provision of Equipment and Supplies. The RPO should be responsible for ensuring that equipment and supplies appropriate for radiation protection

work are available, are maintained in good working order, and are used properly. Written procedures for the use of the equipment should be available and followed.

3.3.3 Surveillance and Monitoring Programs

Another component of a radiation protection program is surveillance and monitoring, which help keep radiation exposures to personnel and the public ALARA and within applicable dose-equivalent limits. Routine survey programs, used to assess the radiological status of a facility, are discussed in Chapter 4 of this manual. Procedures for monitoring personnel are described in Chapters 5 and 6.

3.3.4 Radiological Design

The terms facility design, radiological design, and radiological engineering are often used interchangeably, although their meanings are different. Facility design refers to a plan for a building or installation as a whole, and thus includes nonradiological as well as radiological design features. Radiological design refers to the specific set of features required in a facility because of the planned presence of radioactive source or radiation-generating machines. Radiological engineering refers to the actual construction of a facility in which radioactive materials will be stored or used. (The term can also be used in a broader context to include design.) Design implies the development of an idea as opposed to the actual construction and operation of a facility.

Proper facility design is an effective approach to reducing occupational exposures. Well-designed facilities provide a greater degree of safety than can be obtained by dependence on administrative rules and procedures alone. Although design can never eliminate the possibility of accidental radiation exposure or contamination, it can reduce the probability and magnitude of such accidents. A qualified expert should therefore participate in the planning and design of new facilities and of modifications to existing facilities. Topics that should be considered in radiological design are discussed in Chapter 8.

3.3.5 Radioactive-Material Control and Waste Management

Proper control of radioactive materials is necessary to ensure that personnel and the general public are protected from unnecessary exposure to radiation. Such control extends to all aspects of radioactive-materials handling, including procurement, use, storage, shipment, and waste disposal.

The RPO should review all procurement and transfer requests for radiation-producing sources and devices and should monitor and inventory radioactive materials when they are received to ensure that they have not been damaged in transit or caused contamination of personnel and facilities. Radiation sources may then be transferred to authorized users in the organization or stored in specially designated facilities until needed. Later transfer of radioactive materials may require special procedures to assure proper controls, and care should be taken to ensure that the person or organization receiving the materials is licensed and authorized to received and use them.

An inventory should be maintained to ensure that the RPO can at any time determine the identity, quantity, and location of all radioactive materials. The location, safe condition, and use of radioactive materials should be confirmed by a periodic audit and by routine surveys performed by the RPO.

The RPO should review the disposal of all radioactive materials. They should be disposed of by transfer in suitably prepared containers to authorized locations for radioactive waste disposal. Transportation is discussed more fully in Chapter 9, radioactive-waste disposal in Chapter 10, and inventory record systems in Chapter 13.

3.3.6 Emergency Planning

Every facility in which radioactive material, radiation-generating devices, or radiation sources are produced, used, or stored should have an emergency plan. The emergency plan may be simple or complex, depending upon the facility. In all cases, however, it should be documented, reviewed periodically, and tested at least yearly.

An emergency plan is created through evaluation of the accident potential of a facility. The emergency actions necessary to reduce the consequences of potential accidents, and the individuals responsible for those actions, are

then determined. Coordination with outside emergency forces (public information officials, hospitals, and police, fire, and health departments) is also planned. When an emergency plan has been established, realistic exercises in which key staff members participate should be held to test the adequacy of emergency preparedness. These exercises should include tests of evacuation procedures, the use of emergency equipment, and those rescue and first aid techniques in which staff members may play a role. Periodic testing of emergency equipment and instrumentation is also necessary. Procedures for developing a plan are described in Chapter 11.

3.3.7 Personnel Selection, Qualification, and Training

Adequate training is fundamental to a radiation protection program. Appropriate training should be extended to the radiation protection staff, installation management, and radiation workers. Training programs are discussed in Chapter 12.

3.3.8 Recordkeeping

Documentation is needed as evidence to support the reliability and effectiveness of a radiation protection program. Records should be complete and should reveal the patterns of radiation exposure at the facility. Data on all operating and working conditions should also be available. The records that should be considered for retention are described in Chapter 13.

3.3.9 Quality Assurance Program

A quality assurance program is a means of verifying that each part of a radiation protection program is being carried out adequately and that the total program meets its purpose. A quality assurance program should be developed for any facilities or locations where the following take place:

1. radioactive material is received, used, stored, or prepared for disposal
2. radiation-generating machines are operated
3. personnel dosimeters are evaluated
4. radiation detection or measuring equipment is procured, received, repaired, calibrated, or used

5. facilities or equipment that will be used for these activities are designed, constructed, or modified.

Quality assurance is discussed in Chapter 14.

Section 3.4 ADMINISTRATION OF THE RADIATION PROTECTION PROGRAM

The success of a radiation protection program is dependent on firm management commitment to the program and on the availability of individuals who have a thorough understanding of radiation protection principles. Within the DA, the overall responsibility for the radiation protection program rests with the commander, director, or chief of the installation or activity. The management and administration of the radiation protection program is delegated to designated personnel such as the RPO or the IRCC. The IRCC is an advisory body that assists the commander in establishing local rules and procedures for the procurement, storage, and safe use of radiation sources. The committee consists of the commander, the RPO, the safety officer, and the medical officer (if available)--or representatives of these individuals--together with a representative of employee groups, and others knowledgeable in radiation protection. The RPO is generally responsible for the implementation of the radiation protection program. This individual must be technically qualified through education, training, and professional experience.

The assignment of responsibility must be accompanied by accountability and authority. Authority granted to the radiation protection staff should be broad and fully supported by upper management. Specific authorities should include the following:

1. approve plans for the construction or modification of facilities in which radioactive materials will be used or stored, or in which radiation-generating machines will be located
2. issue and approve standing operating procedures or job safety analyses (this implies review and approval of operating plans and procedures before their implementation)

3. determine operational protective measures to ensure that exposures are kept ALARA

4. train and assess the qualification of radiation workers

5. plan for and establish equipment and procedures for monitoring and control of personnel exposures.

Authorities should be delineated between the IRCC and the RPO. One way this can be done is outlined below.

3.4.1 Ionizing Radiation Control Committee

The duties of the IRCC can include the following:

1. Review proposals for the use of ionizing radiation sources and recommend protective measures to the commander (AR 40-14).

2. Prescribe any special conditions and requirements that may be needed (such as physical examinations, additional training, designation of limited areas or locations of use, disposal methods, etc.).

3. Prepare and disseminate information on radiation safety for use by and guidance of personnel.

4. Pass judgment on the adequacy of safety measures and health protection for safeguarding personnel.

5. Keep a record of actions taken in approving the use of radioisotopes, and of other transactions, communications, and reports involved in the work of the committee.

6. Provide policy direction to the RPO, based upon state and federal regulations and licenses, for the use of ionizing radiation at the installation.

7. Approve or disapprove all applications from prospective users of radioactive materials and from prospective operators of sources of ionizing radiation.

8. Approve or disapprove all applications for laboratories in which radioactive materials would be used or in which sources of ionizing radiation would be operated.

9. Review plans for all new buildings or for modifications to existing buildings in which radioactive materials or other sources of ionizing radiation would be used.

10. Suspend any operation that, in the opinion of the IRCC, represents a serious radiation hazard or violates applicable regulations.

3.4.2 Radiation Protection Officer

The RPO's responsibilities can include the following:

1. Ensure compliance with current directives for radiation protection.

2. Provide consultation on the hazards associated with radiation and the effectiveness of measures to control these hazards.

3. Supervise the radiation protection program and advise on the control of hazards to health and safety.

4. Coordinate the day-to-day administration and development of the radiation protection program.

5. Disseminate information on radiation safety and health physics.

6. Review all proposals for radiation usage and recommend to the IRCC approval or disapproval of all applications from prospective users of radioactive materials and from prospective operators of sources of ionizing radiation. Detailed information on such reviews is given in Appendix A.

7. Inspect facilities and equipment on behalf of the IRCC.

8. Review plans for all new radioisotope and radiation facilities.

9. Obtain all necessary licenses and registrations pertaining to radioactive materials and sources of ionizing radiation for the installation or activity.

10. Develop procedures for the purchase and transfer of radioactive materials.

11. Develop procedures for the disposal of solid and liquid radioactive wastes.

12. Maintain required records, including the following: personnel dosimetry, radioactive waste disposal, radioisotope inventory, instrument calibration, and leak tests on sealed sources.

13. Provide radiation surveys and monitoring of all radioisotope and radiation facilities.

14. Offer brief courses on radiation safety for users and prospective users of radioactive materials and ionizing radiation.

15. Suspend any operation that, in the opinion of the RPO, represents a serious radiation hazard or violates applicable regulations. The operation suspended will be reviewed by the IRCC.

REFERENCES

National Council on Radiation Protection and Measurements (NCRP). 1971. Basic Radiation Protection Criteria. NCRP 39, Washington, D.C.

National Council on Radiation Protection and Measurements (NCRP). 1977. Review of NCRP Radiation Dose Limit for Embryo and Fetus in Occupationally Exposed Women. NCRP 53, Washington, D.C.

U.S. Code of Federal Regulations. 1982. Title 10, Part 19, "Notices, Instructions and Reports to Workers; Inspections." U.S. Government Printing Office, Washington, D.C.

U.S. Code of Federal Regulations. 1982. Title 10, Part 20, "Standards for Protection Against Radiation." U.S. Government Printing Office, Washington, D.C.

U.S. Code of Federal Regulations. 1982. Title 10, Part 40, "Domestic Licensing of Source Material." U.S. Government Printing Office, Washington, D.C.

U.S. Code of Federal Regulations. 1982. Title 21, "Food and Drugs." U.S. Government Printing Office, Washington, D.C.

U.S. Code of Federal Regulations. 1982. Title 39, "Postal Service." U.S. Government Printing Office, Washington, D.C.

U.S. Code of Federal Regulations. 1982. Title 40, "Protection of Environment." U.S. Government Printing Office, Washington, D.C.

U.S. Code of Federal Regulations. 1982. Title 49, "Transportation." U.S. Government Printing Office, Washington, D.C.

U.S. Department of Defense, Defense Supply Agency. Radioactive Commodities in the DOD Supply Systems. DSAM 4145.8, AR 700-64, Washington, D.C.

U.S. Department of the Army, Headquarters. Safety - Ionizing Radiation Protection (Licensing, Control, Transportation, Disposal, and Radiation Safety). AR 385-11, Washington, D.C.

U.S. Department of the Army, Headquarters, Army Materiel Command. Safety - Radiation Protection. DARCOM-R 385-25, Washington, D.C.

U.S. Department of the Army and Defense Logistics Agency. Medical Services - Control and Recording Procedures for Exposure to Ionizing Radiation and Radioactive Materials. AR 40-14, DLAR 1000.28, Washington, D.C.

APPENDIX A

REVIEW OF PROPOSALS FOR RADIATION USE

APPENDIX A

REVIEW OF PROPOSALS FOR RADIATION USE

The RPO is responsible for reviewing project plans, personnel, and facilities before work with radioactive material or radiation-producing devices is begun. Standing operating procedures are then prepared by the RPO and the IRCC, and records are maintained as the work proceeds.

A.1 PROJECT EVALUATION

The project should be evaluated based on the following criteria:

A. License. The RPO should check the site or facility license to ensure that the radioactive material proposed for use can be brought onsite and that the proposed chemical and physical form and the proposed uses of the material are allowed by the license. If the license does not show the proposed uses, an amendment to the license must be requested. For assistance in preparing an amendment to the license, or in interpreting the license to determine whether an amendment is necessary, contact DARCOM headquarters.

Six months or more may elapse before a requested license amendment is authorized. Project leaders should be made aware of the possibility of delay; they can then inform the RPO of the needs of the project early enough in the planning process so that the license amendment will be approved at about the same time as the project is scheduled to begin.

B. Radionuclide. The RPO should assess the radionuclide to be used, considering whether an alternate, less hazardous radionuclide could be used (for example, ^{33}P rather than ^{32}P) and whether a radionuclide is necessary at all or whether other methods of achieving the same purpose are available.

C. Quantity. The amount of radioactive material used for the project should be the minimum possible. If possible, the stock quantity of radioactive material should be divided into small aliquots.

The quantity of radionuclide proposed for use should be compared with the amount for which the site or facility is licensed. This comparison should take into account both the quantity to be used during the project and the total onsite inventory for that radionuclide. The inventory of concern includes both quantities in laboratories and waste quantities that are stored and waiting for shipment.

D. Chemical and Physical Form. If the material proposed for use is volatile, the need for a volatile form should be assessed. Chemical methods for reducing the volatility of the chemical compound may be available; for example, raising the pH of an iodine solution reduces the amount of iodine released into the atmosphere. Concentrated solutions of alpha-emitting radionuclides, such as ^{244}Cm, may present difficulties. Dilute solutions are less likely to cause volatilization.

E. Work Procedures. The RPO should consider whether there are standard procedures for doing the proposed work; whether the proposed work follows the established procedures; whether the procedures can be improved, for example, by reducing the work time; and what types of protective apparel should be worn.

A.2 PERSONNEL CONSIDERATIONS

Personnel considerations in the assessment of a project include:

A. Pregnant or Potentially Pregnant Women. The DA recommends in AR 40-14 that, during the entire gestation period, the maximum dose equivalent to an embryo-fetus from occupational exposure of the expectant mother should not exceed 0.5 rem. Because pregnancy may not be confirmed for two or more months after conception, women staff members should be made aware of this recommendation and should be encouraged to tell the RPO when they are contemplating pregnancy or as soon as pregnancy is suspected.

B. Minors. Individuals under 18 years of age shall not be exposed to more than 10% of the occupational dose limits.

C. Education and Training. Personnel assigned to work on projects involving the use of radioactive materials or radiation-generating sources should be educated as to the hazards associated with radiation and trained in

the specific skills required for their job. Their attendance at education and training sessions should be documented by attendance rolls, and the RPO should administer tests that cover the material following the training sessions. The tests should show whether the material was understood and indicate areas of training that require increased emphasis.

D. Personnel Monitoring. The RPO should ensure that personnel who will work with radioactive materials are provided with appropriate monitoring devices. Monitoring devices such as film badges shall be worn by all personnel who receive, or may be expected to receive, a radiation dose higher than 5% of the applicable standard to the whole body or skin. In practice, whole body badges are usually issued to all individuals who work with x- or gamma-ray sources or with beta emitters that have a maximum energy of 1.0 MeV. Film badges should also be worn by individuals who work around particle accelerators and neutron sources.

Extremity monitors should be worn by individuals who may receive an extremity dose higher than 5% of the applicable standard.

A.3 EVALUATION OF FACILITIES

The facility or work area in which the project will be carried out should be evaluated to ensure that radioactive materials can be used safely. The U.S. Environmental Hygiene Agency and DARCOM headquarters should be contacted for assistance. The information to be considered includes:

A. Shielding. The amount of shielding required depends on the radionuclide to be used (or the operating energy of the radiation-producing machine), the quantity of radioactive material to be present (or the operating time of the machine), and the proposed use of adjacent areas. If shielding already exists, the RPO should assess whether it will be sufficient, how much additional shielding will be required, and whether the building can support the required shielding.

B. Equipment. The working area should have appropriate equipment, which may include hoods, glove boxes, and air filter systems.

A.4 STANDING OPERATING PROCEDURES

After the project has been analyzed, a standing operating procedure (SOP) is prepared by the IRCC and the RPO. The SOP is a summary of the safety findings and a listing of the procedures that must be followed during the course of the project. The SOP should include the following items:

1. type of protective apparel required, if any
2. posting requirements
3. radiation-monitoring devices required
4. personnel dosimeters required
5. bioassay types and frequency
6. recordkeeping requirements
7. reiteration of applicable administrative guidelines
8. any special procedures that may be required.

A.5 RECORDKEEPING

The purpose of recordkeeping is to help the RPO 1) document the radiation doses received by personnel and 2) assess trends in the rate at which doses are being received over time. Recordkeeping also allows the RPO to compare the doses received by staff members who are working on similar projects and in this way to learn which techniques result in the lowest doses to workers. It can also make possible intercomparisons of doses received during similar projects at different facilities.

CHAPTER 4. RADIATION SURVEY PROGRAMS

CHAPTER 4. RADIATION SURVEY PROGRAMS

Routine survey programs are used to evaluate actual or potential radiation hazards at facilities where radiation sources are used. Surveying and monitoring are ways of maintaining radiation exposure to personnel and the environment at a level that is as low as is reasonably achievable (ALARA) within applicable dose-equivalent limits.

The terms "radiation survey" and "radiation monitoring," although frequently used interchangeably, are not synonymous. A radiation survey is an evaluation, under specific conditions, of the radiation hazard associated with the production, use, or storage of radioactive materials or other sources of radiation. Radiation surveys are conducted both in the working environment and in the environment surrounding a facility. Radiation monitoring, an activity frequently performed during a survey, is the measurement of radiation fields or radioactive contamination using fixed or portable instruments. Radioactive contamination can be defined as any radioactive material that has escaped from its intended location or container, or as the deposition of radioactive material in any place where it is not desired, and particularly in any place where its presence might be harmful. Radioactive contamination can be any combination of alpha-, beta-, gamma-, or neutron-emitting radionuclides. Radiation surveys and radiation monitoring are usually performed by the Radiation Protection Officer (RPO) or a member of the radiation protection staff.

Survey requirements and procedures for facilities where radiation sources or radioactive materials are produced, used, or stored are discussed in this chapter. Specific radiation monitoring procedures are also described, as are special requirements for facilities that house nonmedical x-ray units. The objectives and development of environmental survey programs are discussed briefly at the end of the chapter.

Section 4.1 SURVEY REQUIREMENTS

Radiation surveys are recommended or required for certain types of facilities and for specific areas within those facilities. The frequency of surveys varies depending on the facility, area, and other factors.

4.1.1 Facilities That Require a Survey Program

A routine survey program should be considered for any facility where the radiation level may be higher than the natural background level. A survey program is required for facilities that contain the following specific sources:

1. radioactive solids that exceed 1 μCi in activity, that have a specific radioactivity exceeding 0.002 μCi/g, or that emit radiation at a dose rate of 0.1 mrad/hr or more at contact
2. materials controlled by the Nuclear Regulatory Commission (NRC), in quantities that exceed those listed in Title 10 of the U.S. Code of Federal Regulations, Section 30.71, Schedule B (10 CFR 30.71)
3. machines that produce radiation, for example, x-ray devices, accelerators, and electron microscopes
4. radioactive gases or liquids in concentrations that exceed the values listed in 10 CFR 20, Appendix B, Table II
5. items activated in nuclear reactors, by accelerators, or by nuclear weapons.

4.1.2 Areas Within a Facility That Require a Survey Program

Facilities are generally divided into a series of sequential areas according to the radiation hazard in each area. The designations of these areas helps control personnel exposure to radiation. The areas used are: 1) radiation areas, 2) controlled areas, and 3) uncontrolled areas. Each of these areas should be surveyed by a member of the radiation protection staff. The areas are described briefly below and more fully in Chapter 8, "Selection of Radiation Facilities."

A. Radiation Areas. Radiation areas include three subclassifications: radiation areas, high-radiation areas, and airborne-radioactivity areas.

A radiation area is defined in 10 CFR 20 as any area accessible to personnel in which radiation levels could result in a major portion of the body receiving a dose in excess of 5 mrem in any 1 hour or 100 mrem in any 5 consecutive days. Practically, this would be any area in which the dose-equivalent rate is greater than 2 mrem/hr but less than 100 mrem/hr. A high-radiation area is any area accessible to personnel in which radiation levels could result in a major portion of the body receiving a dose in excess of 100 mrem in any 1 hour. An airborne-radioactivity area is any area, enclosure, or operating area in which airborne radioactivity exceeds the concentrations specified in 10 CFR 20, Appendix B, Table 1, Column 1 or in which the concentration of airborne radionuclides, averaged over the number of hours an individual works, will exceed 25% of the amounts specified in 10 CFR 20, Appendix B, Table 1, Column 1.

B. Controlled Areas. Controlled areas are areas controlled for the purpose of protecting personnel from exposure to radiation. Normally, they are areas adjacent to radiation areas. They are usually free of contamination, but they could become contaminated because of accidental spreads or releases from the radiation area or because radionuclides and contaminated equipment may be transported through them.

C. Uncontrolled Areas. Uncontrolled areas are areas where direct radiation exposure is not necessary or anticipated in the performance of a job. These areas include "cold" laboratories (those containing no activity), offices, lunchrooms, conference rooms, and reception areas. Access to these areas does not need to be restricted for radiological reasons.

4.1.3 Frequency of Surveys

Radiation areas, high-radiation areas, and airborne-radioactivity areas should be surveyed at least once each month. Permanent storage areas may be exempted from monthly surveys at the discretion of the Ionizing Radiation Control Committee (IRCC). However, the time between surveys of storage areas may not exceed 12 months. Controlled areas should be surveyed on a routine basis.

The frequency of surveys should increase if changes in conditions or procedures could increase the possibility of personnel exposure. Daily surveys

or continuous monitoring may be required if conditions are highly variable or unpredictable, if unsealed radioactive materials are being handled directly, or if a radiation accident has occurred.

Surveys should be conducted before an operation involving radiation sources is begun and before changes in an existing operation are approved. A survey is also required at the termination of a project involving the use of radiation, to verify that no contamination exists and that radiation sources and radioactive materials have been properly stored or disposed of.

All sealed sources in quantities larger than the quantities listed in 10 CFR 30.71, Schedule B, must be leak tested at least every 6 months, unless specifically exempted by a DA authorization or an NRC license. Alpha sources in quantities larger than those listed must be tested every 3 months, unless otherwise exempted.

Section 4.2 ROUTINE SURVEY PROCEDURES

An effective routine survey program includes the following steps: 1) preparation, 2) inspection and measurement, 3) evaluation and recommendations, and 4) completion of records and reports. These steps are described in detail below. Special considerations for the survey of facilities containing nonmedical x-ray devices are considered in Section 4.4.4.

4.2.1 Preparation

It is essential that adequate preparation be made before any routine survey is conducted. The member of the radiation protection staff who is conducting the survey must be thoroughly familiar with the sources of radiation and the nature and purpose of the work performed in the facility. The steps for complete preparation are: 1) gathering information, 2) diagramming the installation, 3) preparing an inspection list, and 4) obtaining necessary equipment and material.

A. Gathering Information. Preparation for a survey should begin with the gathering of information about the radiation sources present, their intended use, and the physical safeguards and written procedural controls used to

minimize personnel exposure to radiation. This information can be obtained by talking to personnel and by examining plans, drawings, records, and written procedures. A file containing all information pertinent to a particular facility or work area should be maintained. Examples of the types of information to be obtained and filed are:

1. the types and numbers of sources used (e.g., sealed sources, unsealed sources, or radiation-generating devices)
2. the types and energies of radiation produced by the sources, together with any information about absorbers or moderators used to alter the initial energy spectra
3. the geometry, size, and position of radiation fields
4. the direction of beams produced by radiation-generating devices
5. the chemical composition and physical form of radioactive materials
6. the expected type(s) of radiation and/or contamination (e.g., alpha, beta, gamma, neutron)
7. the potential for release or dispersion of radioactive material
8. the procedures and the nature of the facilities used for the storage, handling, transportation, and disposal of radiation sources and radioactive material
9. the design and construction of devices for containing unsealed radioactive materials and sources (e.g., hoods or glove boxes)
10. the design of ventilation and exhaust systems
11. the design of interlock, alarm, and emergency shutdown systems
12. the nature of fixed monitoring equipment used in the facility
13. the locations inside and outside the facility that are occupied by personnel, and whether persons potentially exposed there are classified as occupationally or nonoccupationally exposed
14. protective barriers used for exposure control
15. standing operating procedures (SOPs)
16. previous survey records

17. emergency plans

18. the training and experience of personnel working with the radiation sources.

B. Diagramming the Facility. The second step in preparing for a survey is to make a diagram of the facility showing the location of radiation areas, controlled areas, and uncontrolled areas. The relative position of sources, work areas, waste storage areas, and disposal areas within radiation areas should also be shown. Such a diagram can be useful in identifying locations where radiation measurements should be made. The location of the following items should be included on the diagram when appropriate:

1. radiation sources, radiation-generating devices, and radioactive materials
2. the direction of beams produced by radiation-generating devices
3. radiation areas, controlled areas, and uncontrolled areas
4. protective barriers (e.g., ropes, shielding)
5. interlocks, alarms, emergency shutdown systems, and warning signs
6. equipment, such as hoods and glove boxes, used to contain unsealed radioactive sources and materials
7. waste storage and disposal areas
8. ventilation and exhaust systems
9. monitoring equipment.

C. Preparing an Inspection List. After reviewing all the information related to the facility, the radiation protection staff member conducting the survey should list all the items to be inspected during the survey. The inspection should include a review of the adequacy of procedural controls and physical safeguards used to control personnel exposure, and verification that all radiation protection procedures are being complied with. A review of the lists above can be useful in preparing the inspection list. Examples of items that could be included are:

1. the presence, location, use, and physical integrity of each radiation source

2. the means of identifying each radiation source (e.g., serial number, type, activity, size, room location)
3. the presence and adequacy of required protective barriers (e.g., ropes, shielding)
4. the possibility of inadvertent movement or removal of shields
5. the possibility of change in the orientation of beams produced by radiation-generating devices, or of any change in the position of sources
6. the availability, condition, and use of safety and special-handling equipment (e.g., portable shields, remote-control devices, hoods, protective clothing, showers)
7. the possibility of the introduction of radioactive materials into the facility's effluent stream because of improper air flow or water drainage
8. the adequacy of facilities and procedures for retaining and/or disposing of radioactive waste
9. the facility's design, including traffic flow, any restriction of access or exits, ventilation, the type of surface finish, the location and type of water outlets, and the accessibility of shutoff valves or switches for air conditioning, electricity, water, gas, etc.
10. the presence, correct functioning, and use of protective devices (e.g., interlocks, warnings devices, evacuation alarms, ventilation failure alarms, emergency shutoff switches)
11. the possibility of bypassing protective devices without adequate warning
12. the posting of radiation areas
13. the correct labeling of radioactive materials and radiation sources
14. the adequacy of and compliance with procedures for controlling personnel radiation exposure and for controlling the spread of contamination during the handling, storage, transportation, and disposal of radioactive sources

15. the availability, adequacy, and correct functioning, calibration, and use of survey and monitoring equipment

16. the adequacy of and compliance with routine survey and monitoring procedures

17. the existence, adequacy, and display of emergency plans and the familiarity of personnel with these plans

18. the status of personnel radiation protection training.

D. Obtaining Equipment and Material. After evaluating what type of radiation and/or contamination (alpha, beta, gamma, neutron) can be expected, the surveyor should decide what radiation detection and measurement equipment is needed. The information in Chapter 2 of this manual, "Radiation Instrumentation," is useful for this determination. Other miscellaneous equipment and materials may be needed, for example, clipboards, survey report forms, smears, protective clothing, shoe covers, and disposable plastic gloves.

4.2.2 Inspection and Measurement

When adequate preparation has been made, the inspection can be started and measurements made. The radiation protection staff member who is responsible for conducting the inspection and making radiation measurements should be aware of the controls needed to ensure that his/her own radiation exposure is kept ALARA. Personnel dosimeters, protective clothing, and respiratory equipment should be used when appropriate, and the surveyor should ensure that radiation generators, source-shielding mechanisms, or source-handling equipment cannot be operated except under his/her control during the survey.

A. Inspection. The inspection of a facility is conducted to: 1) provide firsthand knowledge of the installation, personnel, surroundings, radiation sources, and equipment; 2) assess where radiation measurements should be made; and 3) assess the presence and effectiveness of each physical safeguard and the extent of compliance with procedural controls used for radiation protection. The checklist prepared prior to the start of the survey should be useful in identifying the items to be inspected. The surveyor should be alert for any deviation from written plans and procedures.

B. Measurements. The places identified for measurements during the facility inspection should be monitored for contamination, and measurements of the radiation field produced by sources should then be made. Specific monitoring procedures are described later in this chapter.

4.2.3 Evaluation and Recommendations

When all inspections and measurements have been made, the results should be evaluated to determine the overall radiological status of the facility. The evaluation should include a determination of any significant levels of contamination and any significant dose rates produced by sources, and the identification of any deficiencies in the radiation protection program. Recommendations for corrective action should be made so that dose equivalents are kept ALARA. Such recommendations may include changes in:

1. operational factors (e.g., time spent by personnel in radiation areas, equipment use time, or methods of operation)
2. shielding (e.g., size, thickness, type of material, or location)
3. manipulative equipment, particularly relating to the equipment's speed of operation and the distance of personnel from sources
4. procedural controls, particularly those that eliminate unnecessary personnel exposure or contamination
5. personnel protection or warning devices
6. survey and monitoring procedures
7. personnel monitoring and survey equipment
8. plans of action for accidents or emergencies
9. personnel training.

A resurvey may be needed after corrective action is taken, to ensure that the changes made are effective.

4.2.4 Survey Records and Reports

Records of radiation surveys are needed for assessing the effectiveness of the radiation protection program. They may also be useful in interpreting the

results of personnel monitoring. Survey reports should contain the following information:

1. date and time of the survey
2. general location of the survey (building and room)
3. specific locations and objects where radiation measurements were made
4. purpose of the survey (e.g., leak test of sealed source; routine survey for contamination on floors and other surfaces; or survey to establish dose rates to personnel)
5. identification (type and serial number) of the radiation detection instruments used to perform the survey
6. measurement results and conditions observed (e.g., dose rates and contamination levels)
7. conclusions and recommendations
8. identification of the individual performing the survey.

A facility diagram may be attached directly to the report and used to note the dose rates and contamination levels observed during the survey.

More information on records of surveying and monitoring activities can be found in Chapter 13, "Recordkeeping." The degree of detail included in survey records must be sufficient to make them meaningful after the passage of several years. Records should be kept for at least 5 years.

Section 4.3 SPECIFIC MONITORING PROCEDURES

Procedures for measuring radiation fields and contamination, for leak testing sealed sources, and for personnel monitoring, air monitoring, and tritium monitoring are described below. Information on the instrumentation for these procedures is given in Chapter 2, "Radiation Instrumentation."

4.3.1 Measurements of Radiation Fields

Measurements of radiation fields--the areas around sources that receive radiation from the sources--are made to provide a basis for estimating personnel exposure and for determining the effectiveness of procedures used for radiation protection. The number of measurements to be made depends on how much people move about within a given field and how much the field varies in space and time. If the radiation field is fixed, as in many x-ray installations, few measurements are required. However, if the radiation pattern is variable, such as during the removal of a source from a shielded container, more measurements are required. In the extreme case, it may be necessary to continuously monitor work in progress. The intensity of the radiation should be measured using dose rate instruments in locations occupied by personnel. The measurements should be recorded on a data sheet or on a floor plan corresponding to the area monitored and should be compared with specified limits.

Procedures for calculating external exposure are discussed in Chapter 6. It may be useful, when planning the control of an individual's occupational exposure, to compare short-term measurements in a radiation field with estimates of the dose equivalent that would be received by an individual who worked in that field for extended periods of time. For example, if the maximum dose-equivalent rate for a particular radiation field is 10 mrem/hr, and if an individual worked in that field for 5 hours each week, the expected dose-equivalent rate would be: 10 mrem/hr x 5 hr/wk = 50 mrem/wk. The results of this type of conversion can be compared directly with applicable administrative or regulatory limits.

4.3.2 Measurements of Contamination

Familiarity with the work performed in a radiation area is essential for determining what type of surface contamination is most likely to be present, where it is likely to be, and whether it is likely to be fixed or removable. Fixed, or nonremovable, contamination contributes to external exposure. Removable contamination can enter the body and contribute to internal exposure.

Because removable contamination can be spread and presents an internal hazard, the member of the radiation protection staff who is measuring the contamination must be careful to avoid both exposure to himself and the spread

of contamination. The surveyor should wear adequate protective clothing during the survey, taking care to avoid contamination of hands, clothing, and radiation detection instruments. When only gamma radiation is present, the detection instrument can be entirely covered by a thin plastic material for contamination control. The sensitive areas of the detector must not be covered when alpha radiation is present. Shoe covers, gloves, instruments, and other equipment used during an extensive survey should be monitored periodically during the survey. As soon as the entire survey has been completed, protective clothing should be removed and surveyed for contamination, together with the instruments and equipment used.

Direct measurements using portable instruments can be used to determine the total amount of fixed and removable contamination present. An indirect measurement technique is used to detect removable contamination. These two techniques are described below.

A. Direct Measurements. Any area within a facility where there may be contamination should be systematically monitored with a sensitive detection instrument. During the measurement, the probe should be held close to (within 0.6 cm of) the surface. To prevent instrument contamination and damage, the probe must not contact the surface. The probe should be moved slowly over the surface to allow the instrument time to respond. Instrument readings should be recorded on a data sheet or on a floor plan of the area being monitored.

B. Indirect Measurements. A smear taken from a surface that may be contaminated can be used to monitor for removable contamination. A smear test is considered an indirect measurement of contamination.

To perform a smear test, a floor plan of the facility to be monitored is needed, as well as small pieces of paper, such as filter paper discs, to be used as smears. A smear is taken by wiping a 100-cm^2 portion of the surface to be monitored. The items or areas from which smears are taken are identified on the floor plan. The smear should be removed from the facility being monitored and counted according to specified laboratory procedures.

Care should be taken to avoid touching either the surface being monitored or the contaminated side of the smear, and to keep the probe from touching the smear. Cross-contamination of the smears can be avoided by placing each smear

in an individual envelope immediately after the smear is taken. Smears should be treated as radiation sources and handled according to radiological safety procedures.

C. Action Levels and Reporting. The results of monitoring for both fixed and removable contamination should be compared with the contamination limits given in Appendix A. The actions to be taken if the levels found exceed the limits are also identified in the table.

4.3.3 Leak Testing Sealed Sources

The instruments and supplies needed for leak testing sealed sources are 1) a remote-handling tool, 2) sheets of paper with impermeable backing (or sheets of ordinary paper and sheets of polyethylene film), 3) discs of filter paper that have a high wet strength (for making smears), 4) envelopes, 5) rods of wood, plexiglass, aluminum, or some other material, 6) adhesive tape, and 7) a radiation detection instrument.

Before a leak test is begun, a data sheet should be started that includes a description of the source, the type of leak test to be performed, the date of the leak test, and the name of the person performing the test. Space should be left on the data sheet so that the results of the leak test in μCi and any action taken as a result of the test can be recorded later.

Leak testing should be planned so that the surveyor's exposure is kept to a mimimum. The dose rates at given distances from the source should be calculated so that shielding needs, the length of the remote-handling tool needed, and the time allowable near the source can be determined. A rule of thumb is to plan an operation so that the person performing a test or a series of tests does not receive a whole-body dose in excess of 5 mrem. "Dry runs" can be performed if desired.

It is always a safe procedure to assume that a source is leaking and to assess the physical provisions and operations that would be needed to deal with a contamination incident. Knowing the construction of the source is important so that leak testing does not damage the source. Protective rubber gloves should be worn during the test.

A. Direct Leak Testing. This method is applicable to sealed sources that are not in a container, or that are in a container but are not fastened in it, and that can be handled safely with available ecuipment and facilities. The total whole-body dose received during the test should not exceed 5 mrem. This procedure must be performed in a hood or glove box rather than on an open bench top to prevent possible contamination of the work area.

A sheet of impermeable paper (or paper backed with a polyethylene sheet) should be placed on the working surface and taped down if necessary, to prevent contamination of the working surface if the source is leaking. A clean filter paper disc should be marked to indicate the particular source being leak tested. If the source contains water-soluble radioactive material, the filter paper smear should be dampened with distilled water.

When a contained source has been removed from its shielded container, using the appropriate remote-handling tool and cbserving applicable radiological safety procedures, all of its surfaces should be carefully wiped. The source should then immediately be replaced in its container. Dry smears (or wet smears that have been allowed to dry) shoulc be checked with an instrument that monitors low levels of alpha or beta-gamma radiation, as appropriate. Readings should be taken with the open window of the probe near the smear but not touching it. If contamination is detected, the source is likely to be leaking, and precautionary measures should be taken to avoid unnecessary exposure of personnel until the situation has been fully evaluated. The smear should be counted according to specified laboratory procedures in order to obtain quantitative results.

B. Indirect Leak Testing (Container Interior). This method is applicable to sealed sources that are not in a container, cr that are in a container but are not fastened in it, and that have activity levels that prevent safe direct leak testing with existing equipment and facilities. The test or series of tests should be planned so that the radiation protection staff member performing it does not receive a whole-body dose in excess of 5 mrem.

For this test, a contained source is removed from its normal shielded container and transferred to an alternate shielced container or temporary shielding set up specifically for this purpose. An appropriate monitoring

instrument should be used to ensure that the source in the temporary housing is adequately shielded. In addition, instruments for monitoring low-range beta-gamma or alpha radiation should be used to monitor accessible surfaces of the empty container. Any positive readings should be recorded, and if contamination is detected, further precautionary measures should be taken before the leak test is continued.

For this test, smears of the inside surfaces of the empty source container are taken, particularly of areas normally in direct contact with the source. The smearing device should consist of a rod (of wood, plexiglass, aluminum, or other material) long enough to reach the area to be wiped, with a filter paper smear attached to one end. If the source contains water-soluble radioactive material, the filter paper should be moistened with distilled water. The wet or dry smear should be rubbed on the inside surfaces of the empty container, especially on the surfaces that most closely contact the source. Dry smears, or dried wet smears, should be checked with a low-range beta-gamma or alpha-monitoring instrument, the readings taken with the open window of the probe near the smear. If contamination is detected, steps should be taken to prevent unnecessary exposure of personnel until the situation has been fully evaluated. The smear should be counted according to specified laboratory techniques in order to obtain quantitative measurements.

C. Indirect Leak Testing (Container Exterior). This method is applicable to sealed sources that are fastened in a container. It is also applicable to other sealed sources that cannot be leak tested safely with existing facilities and equipment.

The portions of the shielded container or device where contamination would be expected to appear if the sealed source were leaking should be smeared using the rod-and-smear device described above. All applicable radiological safety procedures should be observed, and the smear should be counted in the same manner as used for the interior indirect leak test.

4.3.4 Personnel Monitoring

Personnel are monitored to determine whether contamination is present on them and to measure internal and external exposure. Personnel monitoring

serves two purposes: 1) to assure that all exposures are maintained ALARA, and 2) to identify any unsuspected source of exposure so that prompt corrective action can be taken.

A. Contamination. Personnel must be monitored for contamination before leaving any area in which radioactive materials or sources are used or stored. If an individual is contaminated, follow-up surveys must be made to determine the source of contamination and to detect any contamination that may have been spread by the individual. Prompt corrective action must be taken to eliminate the source of contamination.

A sensitive detection instrument should be used to monitor personnel. Skin and clothing should be carefully monitored, with an emphasis on the head, hands, and feet. Any point that shows visible signs of contact, such as dirt, grease, or liquid stains, should be monitored. In addition, any surface known to have come in contact with equipment or contaminated surfaces should be monitored.

The probe of the instrument should be held close to the individual's skin or clothing but must not be allowed to contact it. The probe should be moved slowly to allow time for the instrument to respond.

B. Internal Exposure. The principal objective of internal personnel monitoring is to determine whether radionuclides have entered the body. The routine determination of internal contamination is necessary only in facilities where unsealed radioactive materials may become airborne. Internal personnel monitoring should also be considered whenever a routine survey indicates significant levels of contamination.

Internal dose is determined using two indirect methods: 1) radiochemical analysis, which is the measurement of radioactivity in urine, feces, blood, secretions, and body tissues; and 2) in-vivo (or whole-body) counting, which is the measurement of radiation emitted from the body, using an external detector. These procedures are highly specialized. More information on their use and on the control of internal exposures is provided in Chapter 5, "Internal Exposure."

C. External Exposure. The external whole-body dose to an individual is estimated using personnel dosimeters. A personnel dosimeter should be worn by each individual who is occupationally exposed to sources of ionizing radiation. Dosimeters must be worn in radiation areas and should be worn by anyone who periodically enters a controlled area and is likely to receive more than 5% of the quarterly dose-equivalent limit listed in Table 3.2 (Chapter 3). An individual under the age of 18 who enters a controlled area and is likely to receive more than 5% of the quarterly dose-equivalent limit for minors should also use a personnel dosimeter. The dosimeters designated by the DA and other methods of controlling external exposures are described in Chapter 6, "External Exposure."

4.3.5 Air Monitoring

The purpose of air monitoring is to determine the cleanliness of the air in the work area. The need for stringent controls on airborne radioactivity should be stressed in SOPs. High concentrations of airborne radioactive contamination can lead to contamination of surfaces in a facility or the environment, and can result in internal exposure to personnel.

Inhalation is the principal means by which radioactive materials can enter the body. The amount of material deposited in the body depends largely upon the concentration in the air inhaled, the particle size of the contaminant, and the length of time the individual is exposed. Control levels for various isotopes are given in 10 CFR 20, Appendix B, Table I. To determine whether control levels are being met, routine air samples are collected and evaluated.

Criteria for the development of an air monitoring program are given in Chapter 5. Several useful references are included in the bibliography. Equipment used to monitor air is discussed in Chapter 2, "Radiation Instrumentation."

4.3.6 Tritium Monitoring

Tritium is a radioisotope of hydrogen that decays to helium by the emission of a beta particle with a maximum energy of 18 keV and an average energy of 5.7 keV. The weak beta particle has a maximum range of 6 μm in water or

0.5 cm in air. When released to the environment, tritium can enter biological materials by several routes. It can be taken into the body in water, in foods, or as tritium or tritium oxide in inhaled air. In both gaseous and liquid forms, tritium can readily penetrate directly through human skin surfaces. Tritium's ability to be readily incorporated into biological systems makes it of concern from the standpoint of internal exposure.

The low energy of the beta particle emitted by tritium creates a special monitoring problem. Portable detection instruments cannot be used because the distance between the tritium source and the detector is usually greater than the particle's range, and even in detectors with a window, the window may be too thick to be penetrated by the beta particle. Windowless gas-flow proportional counters and liquid scintillation counters are therefore used to assay, or test, for tritium. In the special case of tritium gas, ionization chambers may be used. These instruments are described in Chapter 2, and their application for monitoring tritium levels in water, in urine, on surfaces, and in air is reviewed briefly below. Additional references specific to tritium measurements are provided in the bibliography at the end of the manual.

A. Water. The maximum permissible concentration (MPC) of tritium in drinking water is 3×10^{-9} μCi/ml (10 CFR 20, Appendix B). This MPC corresponds to 110 disintegrations per second in each ml of water (dps/ml). Liquid scintillation counting is the method of choice for measuring tritium in water.

B. Urine. A radioassay for tritium in urine should be performed every 2 weeks for all personnel who routinely work with tritium, and immediately following any unusual occurrence involving the spread of tritium contamination. If tritium is found in urine, additional urine samples should be obtained daily to determine the biological half-life of the tritium deposited in the body. Biological half-lives between 7 and 12 days are commonly observed.

Several hours are needed before tritiated water becomes equally distributed throughout the body. Consequently, urine samples should not be taken immediately after a potential tritium inhalation. Generally, 2 to 4 hours should elapse between the time of the exposure and the time of sample collection. When a urine sample is collected, personnel should remove all protective

clothing and wash their hands to avoid contaminating the sample. The urine sample should be placed in an air-tight container and refrigerated. Liquid scintillation counting is used for the radioassay of tritium in urine.

C. Surfaces. Because the energy of the beta particle emitted by tritium is too low to allow the particle to enter portable detectors, a smear test should be used to monitor for surface contamination. The procedure is similar to that described in Section 4.3.2 except that the smear should be lightly coated with glycerin or moistened with water to increase its efficiency in collecting contamination. Smears should be placed into vials immediately after each sample is taken. The sample can be counted using liquid scintillation.

D. Air. In air, tritium occurs primarily as water vapor or hydrogen gas. Flow-through ionization chambers and proportional counters can be used to monitor air for tritium. Ionization chambers cannot distinguish tritium from some other types of radioactive particles and are sensitive to interference from cigarette smoke, aerosols, and external gamma fields. Gas-flow proportional counters can partially discriminate against other radionuclides and are less sensitive to aerosols. The sensitivity of ion chambers is similar to that of gas-flow proportional counters (in the pCi/cm^3 range). To detect tritium levels much below about 1 pCi/cm^3 in air, it is necessary to remove tritiated water vapor from the air using silica gels and bubblers. Information on this procedure is given in Report No. 47 of the National Council on Radiation Protection and Measurements (NCRP 1976). Liquid scintillation counting can be used to assay the water vapor samples.

Section 4.4 NONMEDICAL X-RAY INSTALLATIONS[a]

X-ray equipment poses a potential hazard, both for those who operate it and for those who may be in the vicinity, because of the extremely high dose rates generated by the devices at the flip of a switch. Extensive engineered safeguards and administrative controls are used to minimize normal operating

(a) For this section of the manual, an installation is defined as the space occupied by a radiation-generating source with its associated equipment.

exposures and prevent accidental exposures. Radiation protection is accomplished through the combined efforts of the manufacturers of the devices, the designers and builders of the installations where the devices are used, the operators of the equipment, and radiation protection personnel.

Requirements for the design and operation of x-ray installations are discussed in two reports of the American National Standards Institute, ANSI N543-1974 and ANSI N537-1976. Installations, including necessary shielding, should be designed by a qualified expert and should meet applicable regulations of federal, state, and local agencies.

This section describes the classification of nonmedical x-ray installations, the engineered and administrative safeguards used in them to minimize exposures, and procedures for surveying them. A discussion of surveys for medical x-ray installations is beyond the scope of this manual; information on this topic can be found in NCRP Report No. 33 (1968).

4.4.1 Classification of Nonmedical X-Ray Installations

Installations are divided into four classes, which are described briefly below and in greater detail in ANSI 543-1974. A separate classification for x-ray diffraction and fluorescence analysis equipment is described in ANSI N43.2-1977.

A. Protective Installation. An x-ray unit within a permanent, shielded enclosure is considered a protective installation if the exposure rate at any accessible surface of the enclosure is less than 0.5 mR/hr during operation of the device. Personnel may not remain inside the enclosure during irradiation.

B. Enclosed Installation. An enclosed installation is similar to a protective installation in that the x-ray unit is within a permanent, shielded enclosure. However, a higher exposure level is allowed for this class of installation. The exposure rate at any accessible, occupied area 30 cm from the outside surface of the enclosure must not exceed 10 mR/hr and the exposure rate at any accessible but normally unoccupied area may not exceed 100 mR/hr. During operation of the device, personnel may not remain inside the enclosure.

C. Unattended Installation. An x-ray unit in a shielded enclosure that is small enough to prohibit personnel occupancy is considered an unattended

installation if the exposure rate 30 cm from the outside surface of the device does not exceed 2 mR/hr during operation of the unit. The shielded enclosure may not be used for any purpose other than to enclose the x-ray unit.

D. Open Installation. An x-ray unit that is not in a shielded enclosure and that is located in an area that may potentially be occupied by personnel during operation of the device is considered an open installation.

4.4.2 Engineered Safeguards

Engineered safeguards are safety systems such as warning devices, shields, and interlocks that are built into either the x-ray installation or the x-ray device itself. They should be designed by a qualified expert in accordance with the requirements of the installation class. The fail-safe principle is used whenever possible in the design and construction of safety systems. A fail-safe system is a system in which any malfunction, including malfunction of the safety system, causes the device to stop functioning or to fail in a manner that does not expose personnel to radiation.

Examples of the engineered safeguards required for each installation class are described below. Greater detail can be found in ANSI N543-1974. Engineered safeguards for x-ray diffraction and fluorescence analysis equipment are described in ANSI N43.2-1977.

A. Protective Installation.

1. Each machine must be totally enclosed within physical barriers that have sufficient shielding to reduce exposure rates during operation to less than 0.5 mR/hr at all points accessible to personnel.
2. All entrances to the installation must have a fail-safe interlock system that prevents inadvertent entry during machine operation.
3. The enclosure must be equipped for emergency exit when the doors are locked from the outside. A least one clearly marked scram button (emergency power-cutoff switch) must be located conspicuously in the exposure room. Enough switches must be installed to allow a person to reach a switch within 5 sec after a warning alarm is activated. The purpose of the scram button must be clearly marked.

4. Fail-safe visible and audible warning signals within the enclosure must be actuated at least 20 sec before irradiation begins. The visible signal must stay on during the entire operation of the equipment. Specifications for audible signals are provided in ANSI N2.3-1967.

5. A steady red light activated by the control circuit must be located outside the entrance to each enclosure. A warning sign showing the radiation symbol and the words "Caution: Entering Radiation Exposure Room" must also be posted.

B. Enclosed Installation. The engineered safeguards for protective installations also apply to enclosed installations with the exception of item 1 pertaining to exposure rates. For enclosed installations, each machine must be totally enclosed within physical barriers that have sufficient shielding to reduce operational exposure rates at all accessible and occupied points to less than 10 mR/hr and at all accessible but normally unoccupied points to less than 100 mR/hr. The following additional safeguards are also necessary:

1. All accessible areas in which the exposure rate exceeds 5 mR/hr must be posted with a sign showing the radiation symbol and the words "Caution: Radiation Area."

2. All entrances to the installation must have a sign posted showing the radiation symbol and the words "Caution: Entering Radiation Area."

C. Unattended Installation.

1. The exposure rate at any accessible area 30 cm from the outside surface of the shielded device may not exceed 2 mR/hr during operation. Service doors to areas with exposure levels higher than 2 mR/hr must be locked.

2. The device must be posted with a sign showing the radiation symbol and the words "Caution: X-Rays."

3. A steady red light that is activated by the control circuit must be installed near the head and beam port(s) of each device.

4. All beam ports that are not in use must be secured in a closed position in a manner that prevents their casual opening.

5. The shielding must be secured in a manner that prevents its casual removal or the exposure of personnel.

D. Open Installation.

1. A steady red light that is activated by the control circuit must be mounted on or near the source of radiation.

2. Steady or flashing red lights activated when the device is operating must be located at the radiation area boundary in sufficient numbers to ensure that at least one is visible from each avenue of approach.

3. The perimeter of any area where the radiation level exceeds 5 mR/hr must be posted with a sign displaying the radiation symbol and the words "Caution: Radiation Area."

4. The radiation source and all exposed objects must be within a conspicuously posted perimeter that limits access to areas where the exposure rate is greater than 100 mR/hr. A sign displaying the radiation symbol and the words "Danger: High-Radiation Area" must be posted at the perimeter of this area. During periods of unattended irradiation, this area must be locked to prevent access.

4.4.3 Administrative Controls

Administrative controls are procedures used to minimize the radiation exposure of operating personnel. These procedures require the cooperation of radiation protection and operations personnel. Enclosed, unattended, and open installations require more extensive administrative controls than protective installations because of their higher potential exposure rate.

A. Training. All individuals who use x-ray equipment must be trained to operate it safely. Information on the content of a training program can be found in NCRP Report No. 61 (1978).

B. Standing Operating Procedures. An SOP should be prepared for each x-ray device. The SOP should be posted where it is easy to see, on or next to the console for the device, and should contain the following information:

1. the class of the installation
2. survey and monitoring requirements
3. a list of all required administrative and engineered safeguards
4. operating procedures
5. emergency procedures
6. a list of trained operators
7. the name of the individual responsible for the device.

C. Operation and Maintenance Logs. The individual responsible for an x-ray device should keep two log books: an operations log and a maintenance log. The operations log should contain a complete description of all work performed with the device. The maintenance log should contain a description of all maintenance work. All log entries should be signed and dated.

D. Radiation Area Requirements. X-ray units must be operated only within a radiation area. When a qualified operator is not present, the area must be locked or else the device must be locked prevent its unauthorized operation. Before using the device, the operator must make sure that only required personnel are present within the area and that any exposure of personnel within the area will be minimal.

4.4.4 Surveys of Nonmedical X-Ray Installations

Surveys of nonmedical x-ray installations should include both physical inspection of the facility and measurement of radiation levels. Each installation should be inspected to verify the current and expected occupancy of all areas surrounding the installation. Devices that affect radiation protection (e.g., audible and visible warning signals, shielding, interlocks, and devices that restrict the positioning of radiation sources) should be inspected to determine whether they are operating properly. Administrative controls for each class of installation should be reviewed.

A. Frequency. All new installations must be surveyed before routine operation is begun. Existing facilities should be surveyed every 6 months or whenever changes in the installation could affect radiation protection procedures.

B. Procedure. The RPO should maintain a list of all engineered and administrative safeguards necessary for the safe operation of each nonmedical x-ray installation. Before beginning a survey of an installation, the RPO or a member of the radiation protection staff should review this list and the general procedures outlined in Section 4.2. The following items should be included on the check list for the inspection:

1. Check for a posted, up-to-date SOP. All operators' names must be listed on the SOP.
2. Check for modifications to the device that may affect any safety system (e.g., shielding, interlocks).
3. With the device operating at full power, check for measurable beams of radiation at all appropriate locations. Measurements should be taken at all points accessible to personnel and in other normally occupied spaces, such as offices not related to machine operation. A strong effort must be made to maintain exposure rates ALARA even if they fall within stated guidelines. Thus, it is important to determine and document any exposure rate that could be reduced by administrative or engineered safeguards.
4. Test all engineered safeguards listed on the SOP, including interlocks, warning lights, alarms, and scram buttons.
5. Verify that the device is operated within a radiation area that is adequately posted.
6. Determine that all operators are trained.

C. Radiation Survey Report. A report of a radiation survey of an installation should include:

1. who conducted the survey and the date of the survey
2. the device and installation being surveyed, identified by suitable means (e.g., serial number, room number, and building number or name)
3. the survey instrument used and the date of its last calibration
4. the potential and current at which an x-ray tube was operated during the survey, and any measured x-ray beams

5. the classification of the installation
6. the location of the x-ray source and the orientation of the x-ray beam in relation to each exposure measurement (a diagram may be useful)
7. a description of all engineered and administrative safeguards along with a verification that they were tested or inspected
8. all deficiencies found during the survey and the corrective action to be taken.

Section 4.5 ENVIRONMENTAL SURVEY PROGRAMS

An environmental survey is a systematic investigation and measurement of radiation levels and radioactive contamination levels in the environment surrounding a facility. The objectives of an environmental survey program include:

1. assessment of the natural radiation and radioactivity levels in the environment before operations begin
2. assessment of the actual or potential exposure of man from the additional radioactive materials or radiation contributed to the environment by the facility, or estimation of the probable upper limits of such exposure
3. determination of the fate of contaminants released to the environment
4. detection of sudden changes and evaluation of long-term trends, which can indicate failure or lack of adequate control in the operation of the facility
5. demonstration of compliance with applicable regulations and legal requirements concerning releases to the environment.

The extent of an environmental survey program depends on several factors, including the nature of the facility, the type and quantity of radionuclides handled, and the potential for the release of radioactivity to the environment.

Environmental surveys should be conducted prior to the initiation of radiological operations at a facility and at least once a year thereafter. More frequent surveys may be needed depending on the scope and nature of the facility's activities. The results of an environmental survey should be used to determine any need to modify controls or operations.

The development of a survey program should include the following general steps:

1. Evaluate the facility as a source of direct radiation and radionuclides, especially the composition, concentrations, release rates, points of release, and physical and chemical forms of the nuclides.
2. Identify the pathways leading to exposure to man, using analytical models, the experience gained at other sites, and preoperational data on local meteorology, hydrology, and population distribution and diet.
3. Select the pathways (e.g., water, food, air) that may be most critical in terms of their contributions to exposure, and determine the critical population groups.
4. Determine the measurements required to provide data for dose assessment for normal and abnormal conditions.
5. Allow for flexibility in the program design. As operational experience is accumulated, other types of measurements or measurement frequencies may be desirable.

Details on establishing and carrying out environmental survey programs can be found in the bibliography.

REFERENCES

American National Standards Institute (ANSI). 1967. Immediate Evacuation Signal for Use in Industrial Installations Where Radiation Exposure May Occur. ANSI N2.3, New York.

American National Standards Institute (ANSI). 1974. General Safety Standard for Installations Using Non-Medical X-Ray and Sealed Gamma-Ray Sources, Energies up to 10 MeV. ANSI N543, New York. Also published in 1975 as National Bureau of Standards Handbook No. 114, Washington, D.C.

American National Standards Institute (ANSI). 1976. Radiological Safety Standards for the Design of Radiographic and Fluoroscopic Industrial X-Ray Equipment. ANSI N537, New York.

American National Standards Institute (ANSI). 1977. Radiation Safety for X-Ray Diffraction and Fluorescence Analysis Equipment. ANSI N43.2, New York. Also published as National Bureau of Standards Handbook No. 111, Washington, D.C.

American National Standards Institute (ANSI). 1978. Control of Radioactive Surface Contamination on Materials, Equipment, and Facilities to be Released for Uncontrolled Use. ANSI 13.12 (Draft), New York.

National Council on Radiation Protection and Measurements (NCRP). 1968. Medical X-Ray and Gamma-Ray Protection for Energies up to 10 MeV - Equipment Design and Use. NCRP 33, Washington D.C.

National Council on Radiation Protection and Measurements (NCRP). 1976. Tritium Measurement Techniques. NCRP 47, Washington, D.C.

National Council on Radiation Protection and Measurements (NCRP). 1978. Radiation Safety Training Criteria for Industrial Radiography. NCRP 61, Washington, D.C.

U.S. Code of Federal Regulations. 1982. Title 10, Part 20, "Standards for Protection Against Radiation." U.S. Government Printing Office, Washington, D.C.

U.S. Code of Federal Regulations. 1982. Title 10, Part 30, "Rules of General Applicability to Licensing of Byproduct Material." U.S. Government Printing Office, Washington, D.C.

U.S. Department of the Army, Headquarters, Army Materiel Command. Safety - Radiation Protection. DARCOM-R 385-25, Washington, D.C.

U.S. Department of the Army, Headquarters. Safety - Ionizing Radiation Protection (Licensing, Control, Transportation, Disposal, and Radiation Safety). AR 385-11, Washington, D.C.

APPENDIX A

MAXIMUM PERMISSIBLE CONTAMINATION LEVELS FOR INANIMATE OBJECTS

APPENDIX A

MAXIMUM PERMISSIBLE CONTAMINATION LEVELS FOR INANIMATE OBJECTS[a]

Contaminated Item	Corrective Action	Maximum Alpha		Maximum Beta	
		Fixed[b] (dpm/100 cm^2)	Removable[c] (dpm/100 cm^2)	Fixed[b] (mrad/hr at 2.5 cm)	Removable[c] (dpm/100 cm^2)
1) Personal clothing, including shoes	Replace, decontaminate, or store until radioactive contamination has decayed if above:	200	None	0.05	None
2) Protective clothing					
a. General	Replace, decontaminate, or store until radioactive contamination has decayed if above:	1000	200	0.02	1000
b. Respirators	Replace, decontaminate, or store until radioactive contamination has decayed if above:	200	None	0.6	None
c. Laundry	Release only to licensed launderer if contaminated	-	-	-	-
3) Work areas and equipment[d]					
a. Uncontrolled	Control and post, then decontaminate if above:	200	30	0.05	100
b. Controlled	Decontaminate (or if decontamination is impossible, fix and then check fixation periodically) if above:				
(1) Areas		1000	200	0.2	400
(2) Hoods		1000	200	2.0	2000
(3) Glove boxes		5000	1000	2.5	5000
(4) Workbench surfaces		1000	200	0.5	400
(5) Other equipment		1000	200	2.0	2000
4) Tools, equipment, containers	Prior to nonradioactive use, decontaminate if above:	200	50	0.25	100
5) Vehicles					
a. Used in controlled areas	Decontaminate (or if decontamination is impossible, fix and then check fixation periodically) if above:	1000	300	0.4	500
b. Used in uncontrolled areas	Decontaminate if above:	500	30	0.25	100

(a) Reference: AMC 385-25 and AR 385-11. (Note: These limits may be changed to reflect those found in ANSI 13.12.)
(b) Measured with a calibrated radiation measurement instrument.
(c) Determined using smears analyzed with a calibrated counting system.
(d) For natural and depleted uranium and for ^{238}U, levels for alpha contamination should be increased by a factor of 5, in accordance with NRC guidelines. If ^{226}Ra is a contaminant, levels for alpha contamination should be reduced by a factor of 2.

CHAPTER 5. INTERNAL EXPOSURE

TABLES

CHAPTER 5. INTERNAL EXPOSURE

Internal radiation exposure is the exposure of the body to radioactive materials deposited in the body. Radioactive materials can enter the body through the inhalation of radioactive dusts, mists, and fumes, the ingestion of contaminated food or water, injection via puncture wounds, or occasionally absorption through the skin or via a wound.

Several methods can be used to control exposure of the body to external radiation (see Chapter 6). However, once radioactive material has entered the body, there is usually no practicable method of reducing the internal radiation exposure or the resultant dose. Moreover, if the radioactive material has a sufficiently long half-life, it may continue to irradiate the individual for the rest of his or her life. Because of these difficulties, the intake of radioactive materials into the body must be limited and programs for monitoring the internal exposure of radiation workers should be followed.

When an intake of radioactive material is detected, estimating the resulting internal radiation dose is difficult for several reasons. First, in most cases the quantity of radioactive material taken into the body is not known. Some procedures for assessing this quantity partially solve this problem. Second, radionuclides tend to accumulate, or concentrate, in specific organs of the body, which then receive a larger radiation dose than do other organs. For example, plutonium, strontium, and radium concentrate in the bone; uranium concentrates in the kidneys or lungs (depending upon its solubility); and iodine concentrates in the thyroid. Third, a fraction of the energy emitted by a radionuclide in an organ is absorbed within that organ, while the remainder of the energy escapes to other tissues of the body or leaves the body. The fraction of energy emitted that results in a dose to any single organ depends on several factors, including the type of radiation emitted, the size and shape of the organ and body of the individual, and the distribution of the radioactive material within the organ or body.

In this chapter, procedures for controlling and monitoring internal exposure and for estimating internal dose are discussed.

Section 5.1 CONTROL OF INTERNAL EXPOSURE

Considerable effort should be expended to prevent any intake of radioactive material through the accidental ingestion of removable surface contamination or the inhalation of airborne contamination. Removable surface contamination is radioactive material that is easily moved from a surface by wiping or dissolution using common solvents. Removable contamination presents an external hazard and, more important, an internal hazard if it is ingested. (Fixed surface contamination, which is bound to an object, presents only an external hazard.) Airborne contamination is radioactive material that has become airborne as a result of normal work procedures, suspension or resuspension of surface contamination, breach of containment, sputtering of heated fluids, or vaporization of volatile compounds. Once airborne, the material may be inhaled by personnel, resulting in an internal radiation dose. Airborne contamination can present an additional external and internal hazard if it settles out of the air onto surfaces as removable contamination.

Because of the internal radiation hazard posed by removable and airborne contamination, every means of preventing the spread of contamination should be used. The following approaches are discussed in this section:

1. the use of design features to limit the movement of airborne contamination and the spreading or resuspension of removable surface contamination
2. routine surveys for surface contamination
3. decontamination of contaminated objects and individuals
4. air-sampling and air-monitoring programs
5. the use of protective apparel
6. administrative guidelines.

5.1.1 Contamination Control Through Design Features

Design features are a key element in contamination control. Of particular importance is the design of a facility's ventilation system. Other design features, such as the elimination of surfaces from which material can be

resuspended (e.g., scaffolding, open rafters, and cable runs), are also important in preventing contamination. Contamination-producing substances should be used only in hoods or glove boxes. Such substances would include heated solutions; volatile substances, such as iodine and mercury; and high-specific-activity solutions of alpha-emitting nuclides, such as ^{244}Cm and ^{90}Sr. See Chapter 8 for a complete discussion of facility design.

5.1.2 Contamination Surveys During the Course of Work

Surveys for surface contamination should be conducted routinely, with the frequency dependent upon the radiotoxicity of the material handled, the quantity used, and the relative ease of spreading the contamination. In areas containing radioactive materials that include more than one level of radiotoxicity (see Chapter 1, Table 1.10), all removable contamination should be assumed to be due to the most highly radiotoxic agent until proven otherwise. Personnel surveys should be conducted periodically during the course of work in a radiation area and must be conducted as each person leaves the area. All surveys should be made using the procedures discussed in Chapter 4.

Detection equipment appropriate for the type of contamination involved should be available. For most nuclides that emit beta-gamma radiation, a Geiger-Mueller (GM) survey meter is suitable. If the area contains low-energy-beta emitters (e.g., ^{14}C, ^{35}C), special survey instrumentation such as a thin-window GM should be used. Alpha-emitting nuclides are best counted with windowless proportional counters or with ZnS crystal scintillation detectors. For additional information on instrumentation, see Chapter 2.

5.1.3 Decontamination of Contaminated Objects and Individuals

All contamination should be cleaned up at the earliest possible time. Contaminated objects should be decontaminated to levels below the maximum permissible levels shown in Appendix A of Chapter 4. When an individual is contaminated, the person responsible for decontamination should be given as much information as possible, including the radionuclide(s) involved and the chemical form(s) of each radionuclide. Often, all that is known is that the contaminant is a beta-gamma emitter or an alpha emitter. In many instances, the exposure may be to mixed radionuclides that emit predominantly beta-gamma or alpha radiations.

Instrumentation used to assess the extent of contamination must be able to detect the radiations in question. Use of the wrong type of instrument can lead to underestimation of hazards or failure to detect any contaminants, and to release of the object or individual without proper decontamination. Decontamination procedures for both personnel and objects are discussed in detail in Chapter 7.

5.1.4 Air-Sampling and Air-Monitoring Programs

Air-sampling and air-monitoring programs have two major purposes: 1) to detect the presence of radioactive dusts, mists, and fumes in the air; and 2) to quantify the amount of radioactive material in the air. Sampling devices are designed simply to collect dusts, mists, or fumes; the radioactivity of the sampled material is quantified at a later time. These devices are useful in identifying the amount and type of airborne radiation to which an individual has been exposed. Monitoring devices, on the other hand, detect radioactive material and usually sound an alarm when a specified limit is exceeded. Monitors are generally not as accurate as samplers; however, they do provide an immediate indication of airborne radiation in the work area.

Continuous monitoring or sampling for airborne particulate radioactivity should be conducted whenever personnel have a significant potential for airborne exposure because of radiological conditions in the work area. Continuous air monitors should have both a visual and an audible alarm. Areas where the potential for personnel exposure exceeds the limits of 10 CFR 20, Appendix B, Table I, shall be provided with an air monitor that is sensitive enough to alarm at $\leq$30 maximum permissible concentration-hours (MPC-hr). (An MPC-hr is a unit that expresses the total MPCs an individual has been exposed to. It is the product of the number of MPCs the individual was exposed to and the number of hours the individual was exposed. For an individual exposed to 2 MPCs for 2 hours, for example, the product would be 4 MPC-hr.)

When a continuous air monitor alarms, the following actions should be taken:

1. Personnel who are not wearing respiratory equipment shall immediately leave the area. However, these individuals shall remain in the general

vicinity and shall be surveyed for contamination by a member of the radiation protection staff.

2. The Radiation Protection Officer (RPO) shall be notified immediately.

3. Personnel who are wearing respiratory equipment may remain in the area to stop operations that might be the source of airborne radioactivity. Other personnel may enter the area only if they are wearing appropriate respiratory equipment and only for the purposes of evaluating the source of airborne radioactivity or stabilizing it. When the source of the immediate problem has been identified and controlled, all personnel shall leave the area.

Air samples shall be taken in all potentially contaminated work locations that are not continuously monitored. These samples shall be analyzed to ensure that personnel are not exposed to levels of airborne radioactivity higher than the levels given in 10 CFR 20, Appendix B, Table I. Sampling devices should be located where they will ensure detection of abnormal concentrations of airborne radioactivity. Examples of good sampling locations include on hood faces and above laboratory benches.

5.1.5 The Use of Protective Apparel

The purpose of protective apparel is to place a barrier between radioactive material and the individual. This barrier has negligible shielding characteristics; that is, it does not effectively attenuate, or reduce the intensity of, the radiation reaching the wearer. Its main purpose is to prevent contamination of the skin of personnel and inhalation of airborne radioactive materials. The two classes of protective apparel discussed in this section are protective clothing, which minimizes the contamination of an individual's skin, and respirators, which minimize the inhalation of airborne radioactive material.

A. Protective Clothing. Protective clothing includes gloves, laboratory coats, coveralls, and shoe covers. All protective clothing for use in radiation areas should be clearly marked and easily identified so that it can be kept separate from other clothing.

Because protective clothing often becomes contaminated, it must be removed carefully so that contamination is not transferred to the wearer's skin or street clothing. In all cases, if protective clothing is ripped or torn while an individual is working with radioactive material, the individual should leave the area immediately. The following discussion includes a brief description of the proper methods for removing the clothing.

(1) Gloves. Gloves should always be worn in radiation areas, particularly for handling sealed and unsealed sources or potentially contaminated objects. The best gloves are both strong enough not to tear and tight enough not to continually slip off or catch on experimental apparatus. Disposable surgical gloves are frequently used. "One size fits all" gloves tend to be large and to slip off the hands, and may promote the spread of contamination because of the unconscious movements used to keep them on. In some instances, for example during work with radioactive elemental iodine or alpha-emitting radionuclides, two pairs of gloves should be worn.

Glove removal can cause contamination if not performed properly. During the removal process, avoid quick movements that may cause dust to become airborne. Touch the outside of gloves only with gloved hands, and touch uncontaminated skin only with ungloved hands. Grasp the upper, inside wrist cuff of one glove with the opposite gloved hand and pull down on the glove so that, as it is being removed, it is also being turned inside out. When the first glove is off, it should be held, inside out, in the gloved hand. To remove the second glove, slide the fingers of the ungloved hand down the inside of the gloved wrist until the fingers can grasp the inside cuff of the glove. Grasp the inside cuff with the bare fingers and pull down on the cuff while withdrawing the hand from the glove; this should cause the glove to be turned inside out. Pull the second glove over the previously removed glove. The result should be two inside-out gloves, one inside the other, which are disposed of as radioactive waste. The wearer's hands should be surveyed after the gloves are removed.

(2) Laboratory Coats. Laboratory coats are required for work with radioactive materials. The coats should be correctly sized for the individuals wearing them and should be worn buttoned up. They should be worn only in

radiation areas and controlled areas, where contamination might exist, and should never be worn in uncontrolled areas, where food or beverages will be consumed.

(3) Coveralls. In areas with a high likelihood of contamination or where loose laboratory coats would be inconvenient and might cause excessive resuspension of radioactive materials, coveralls should be used. Coveralls have the relative advantage of protecting all the street clothing of an individual. They can be made of ordinary cloth, special fabrics, or chemically treated papers. Velcro® fasteners make it easier to remove coveralls.

Coveralls are removed as follows. First, remove gloves if they are being worn. Then insert the index finger and the middle finger of each hand inside the front of the collar and loosen the Velcro fasteners by pulling the hands apart. Slide the fingers down the front opening until the coveralls are open below the waist. Place the fingers inside the coveralls at about the height of the collarbone and pull the coveralls off the shoulders and down until the arms are free. Roll the coveralls, inside out, down the body to the ankles, then step out.

(4) Shoe Covers. Shoe covers are required wherever floors may become contaminated. They can be made of any durable material such as plastic, fiber-embedded paper or cloth, or rubber. Shoe covers should be tight enough so that they do not tend to fall off the worker's shoes, but not so tight that they are difficult to remove. A step-off area or pad for removing shoe covers should be located at the exit from the contaminated area. To remove shoe covers, approach but do not stand on the step-off area. Lift one foot so that it crosses in front of the opposite leg, grasp the outside of the cover at the heel with a gloved hand, and pull it off the street shoe, being careful to maintain balance. Do not remove the street shoe with the shoe cover. Place the contaminated shoe cover in a receptacle, then step onto the step-off pad with the street shoe. Do not step on the pad with the remaining contaminated shoe cover. Remove the remaining shoe cover as described above.

® A trademark of Velcro U.S.A. Incorporated.

If the protective gloves have already been removed, shoe covers can be removed by placing the index finger and middle finger of the left hand between the right street shoe and its shoe cover at the inside of the heel, pushing down until the heel of the street shoe is out of the shoe cover, and then sliding the rest of the street shoe out of the shoe cover and placing the right street shoe on the step-off pad. Reverse the procedure for the left foot.

(5) Care of Contaminated Clothing. Contaminated clothing should be placed in receptacles specifically designed for contaminated apparel, and should be sent only to laundries that are equipped to handle contaminated clothing. If protective clothing is worn many times before laundering, it should be stored so that any contamination on it could not be transferred to other items of apparel. Protective clothing contaminated with more than 50 mrad/hr of beta-gamma radiation or more than 40,000 dpm of alpha radiation shall be considered contaminated waste and shall be removed from service.

B. Respirators. Respirators are devices designed to keep the wearer from inhaling airborne radioactive material. Some devices also protect against oxygen-deficient atmospheres. They are not a substitute for either good ALARA (as low as is reasonably achievable) or good engineering practices. Respirators are considered an acceptable method of protecting the health of personnel only under the following circumstances:

1. when the Ionizing Radiation Control Committee (IRCC) has determined that no feasible engineering or work practice controls can be used to control the airborne radioactive material
2. during intermittent, nonroutine operations (1 hour/day for 1 day/week)
3. during interim periods when engineering controls are being designed and/or installed
4. during emergencies.

Respiratory protection programs, the selection of respirators, and the types of respirators available are discussed below.

(1) Respiratory Protection Program. An effective respiratory protection program requires the cooperation of the commander, the RPO, supervisors, and

medical personnel. An adequate program includes, at a minimum, the requirements detailed below. Radiation Protection Officers who are responsible for respiratory protection programs should obtain a copy of the Nuclear Regulatory Commission's NUREG-0041 (NRC 1976) for further detail concerning these requirements.

1. Air sampling and other surveys must be sufficient to identify the radiation hazard, to evaluate individual exposures, and to permit proper selection of respirators.

2. Written standing operating procedures (SOPs) must be followed to ensure proper selection, supervision, and training of personnel using respirators.

3. Written SOPs must be followed to ensure adequate individual fitting of respirators, as well as procedures for testing respirators for operability immediately prior to each use. Individuals who issue respirators shall be provided with training in these procedures.

4. Respirators should be assigned to individuals for their exclusive use, where practicable.

5. Written SOPs must be followed for respirator maintenance (including cleaning and disinfection), decontamination, inspection, repair, and storage. Respirators issued for the exclusive use of one indivudal should be cleaned after each day's use. Respirators used by more than one individual shall be thoroughly cleaned and disinfected after each use.

6. Respirators shall be stored in a convenient and sanitary location. They must be stored where the potential for contamination by airborne or surface radioactive material is minimal.

7. Before initial use, each respirator shall be properly fitted, leakage tests performed, and the facepiece-to-face seal tested in a realistic test situation.

8. Before each use, both positive and negative pressure tests shall be conducted (see Standard Z88.2 of the American National Standards Institute (ANSI 1980)). Respirators shall not be worn when a beard or sideburns, a skull cap that projects under the respirator, temple pieces on corrective

glasses, the absence of one or both dentures, or other conditions prevent a good facepiece-to-face seal.

9. Respirators shall be inspected during cleaning. Experienced personnel shall replace worn or deteriorated parts with parts designed for the respirator. No attempt shall be made to replace components or to make adjustments or repairs beyond the manufacturer's recommendations. Reducing-admission valves or regulators shall be returned to the manufacturer or to a trained technician for adjustment or repair. The manufacturer's parts replacement schedule should be followed.

10. Respirators for emergency use, such as self-contained breathing devices, shall be thoroughly inspected at least once a month and after each use, and a written record kept of inspection dates and findings.

11. Supervisors and personnel shall be instructed and trained in the selection, use, care, and maintenance of respirators. Training shall provide, for each user, an opportunity to handle the respirator, to have it fitted properly, to test its facepiece-to-face seal, to wear it in normal air for a familiarization period, and to wear it in a realistic test atmosphere.

12. Personnel should not be assigned to tasks that require the use of respirators unless the installation's medical authorities have determined that they are physically and psychologically able to perform their work while wearing the prescribed respirator. The medical status of the respirator user should be reviewed periodically, with the frequency of review depending upon the results of appropriate medical examinations, the type of respirator used, and the age of the individual.

13. Bioassays and other surveys should be conducted as appropriate to evaluate individual exposures and to assess the protection actually provided.

(2) Selection of a Respirator. The selection of a respirator depends on a number of health and safety factors, such as the nature of the radiation hazard, the limitations and the intended use of the respirator, how much the respirator limits movement and work rate, the time needed to escape in case of emergency, and training requirements. Because the effectiveness of a respiratory protection program can be determined largely by the degree to which

personnel accept the program, the human factor must also be considered. Personnel acceptance of respirators is influenced by comfort, ability to breathe without undue interference, confidence in facepiece fit, and convincing evidence that a respirator is necessary and that action is being taken where possible to eliminate the need for respirators.

The degree of protection afforded by a given respirator is defined in terms of its protection factor (PF), which is the ratio of the concentration of the contaminant in the ambient atmosphere to that inside the equipment (usually inside the facepiece) under conditions of use. Protection factors are based on laboratory leakage studies and field experience with the device.

Respirators should be selected to provide a PF greater than the multiple by which peak concentrations of radioactive materials are expected to exceed the values specified in Table I, Column I, of 10 CFR 20, Appendix B. For example, if the airborne concentration of a radionuclide in a work area is expected to be five times as high as the permissible concentration listed in the table, then the respirator selected for use in that area should have a PF of 6 or more. The equipment selected should be used so that the average concentration of radioactive material in the air inhaled by the wearer, during any period of uninterrupted use in the area, does not exceed the values specified in the table. For the purpose of this manual, the concentration of radioactive material inhaled when respirators are worn may be estimated initially by dividing the concentration in the air of the work area by the PF. Additional measurements, however, must be taken to evaluate worker exposure.

The protection factors for respirators may not be appropriate where chemical or other respiratory hazards exist in addition to radiation hazards. The selection and use of respirators for such circumstances should take into account recommendations and requirements of the National Institute for Occupational Safety and Health (NIOSH) and the Occupational Safety and Health Administration (OSHA).

The installation's medical authority, or personnel under the guidance of the medical authority, shall determine the type of respirator best suited to each task. The RPO should assist the responsible individual by providing

environmental evaluations and any other appropriate information. Only equipment that is certified by the National Institute for Occupational Safety and Health/Mine Safety and Health Administration (NIOSH/MSHA) should be used.

(3) Description of Respirators. There are basically two forms of respirators: air purifying and air supplying. An air-purifying respirator removes contaminants from the air of the work area by either filtering out particulate contaminants or removing contaminated gases and vapors by chemical means. An air-supplying (or atmosphere-supplying) respirator furnishes respirable air or oxygen to the wearer from an uncontaminated supply.

Respirators are designed to be used with an enclosure such as a facepiece, hood, helmet, or suit. The enclosure excludes contaminated air and ensures that clean, respirable air is supplied to the nostrils and mouth of the wearer.

A facepiece is a tight-fitting enclosure over all or a portion of the face. Only full-facepiece devices should be used to protect against airborne radioactive material. (Facepieces that enclose only a portion of the face are not acceptable for use in radiation areas; they are to be used only for industrial safety applications for protection from nonradioactive particulates, gases, and vapors.) A full-facepiece mask is generally constructed from flexible rubber or plastic and has one or two transparent lenses for viewing. The device completely encloses the wearer's eyes, nose, mouth, and chin. A head harness is attached to the facepiece at five or six points to provide support.

A hood is a loose-fitting, flexible enclosure over the head, neck, and shoulders that is gathered around the neck or shoulders to provide a snug fit. A helmet has a more rigid construction than a hood and protects parts of the head against impacts. Air is supplied to the hood or helmet from a compressed-air supply. Suits are one-piece garments to which a continuous supply of respirable air is provided.

5.1.6 Administrative Guidelines

Some administrative guidelines that will help personnel reduce any intake of radioactive materials are listed below. The list may not be all-inclusive and should not be substituted for common sense in the laboratory.

1. Smoking, eating, and drinking shall not be allowed in radiation areas or controlled areas. The danger of transmitting radionuclides internally is too great.

2. Food containers such as returnable bottles and coffee cups shall not be taken into radiation areas or controlled areas. If they are inadvertently taken in, they should be destroyed.

3. Refrigerators shall not be used to store both food and radioactive materials. Ice cubes from refrigerators used for storing radioactive materials shall not be used for human consumption.

4. Frequently while working with radioactive materials, or upon the completion of work, each individual shall survey hands, shoes, and other areas of the body or clothing that may be contaminated. Contamination should be removed when found and shall be removed before the individual leaves the laboratory. If significant levels of personnel contamination are found, or if the contamination cannot be readily removed, the individual shall contact the RPO.

5. Frequent radiation surveys shall be performed around radiation and/or controlled areas to determine whether there is any deviation from normal background levels of radiation (see Chapter 4).

6. All containers used for radioactive materials shall be labeled in accordance with Army regulations (AR 385-11). Radioactive warning labels, tape, signs, etc., shall not be used for purposes other than those for which they are intended.

7. Radioactive materials shall be stored so that unauthorized individuals are not likely to accidentally handle or otherwise come in contact with them.

8. Each person shall wash hands and arms thoroughly after handling any radioactive source (sealed or unsealed), and in particular before touching any object that goes in the mouth, nose, or eyes.

9. Equipment or apparatus that has come in contact with radioactive materials shall not be used for other purposes until it is demonstrated to be free of contamination.

10. Mechanical devices shall be used for pipetting. NEVER PIPETTE RADIOACTIVE SOLUTIONS BY MOUTH. In addition, to precluce accidental ingestion of radioactive materials through cross-contamiration, mislabeling, etc., never pipette any substance by mouth in labcratories where radioactive materials are used.

11. Radioactive materials in liquid form shall be stored and transported in containers that, if dropped, will not release the materials, for example, in plastic bottles or in glass bottles with styrofoam containers (see Chapter 9).

12. All transfers and dilutions should be performed in functioning exhaust hoods or glove boxes, unless procedures have been approved for working in the open (see Chapter 8).

13. Work should be planned ahead; whenever possible, a dry run to test the procedure should be done first.

14. All items of equipment intended to provide features of safety shall be evaluated periodically to ensure that they are providing the safety feature intended (see Chapter 8). For example, a fume hood in which radioactive materials are handled should provide a uniform air flow through the opening of the hood. This air flow should be checked periodically to ensure that the hood is operating properly.

15. Laboratories shall be kept neat and clean. Equipment or material not being used should be stored away from the work area.

16. Absorbent paper should be placed on work surfaces on which radioactive materials are used. If liquid radioactive materials are used, a container large enough to hold the entire volume of liquid should be positioned to catch any spill.

17. Fingernails should be kept short and clean.

18. If there is a break in the skin below the wrist, gloves of rubber, plastic, or some other substance impervious to the material being worked with shall be worn to cover the break.

Section 5.2 MONITORING INTERNAL EXPOSURE

Inhalation is the pathway by which radioactive material is most likely to enter the body of an occupationally exposed individual. After being inhaled, the radioactive material may be gradually or immediately transferred to the blood, depending upon the solubility of the material, and then excreted from or retained by the body, depending upon other characteristics of the material.

Two methods are used to estimate the amount of radioactive material taken into the body and the consequent radiation dose: radioanalysis and in-vivo counting. Radioanalysis is the measurement of radioactivity in urine, feces, secretions, and other body samples, such as blood and other tissues. In-vivo counting is the measurement of the radiation emitted from the body, using an external detector. Radioanalysis and in-vivo counting are bioassay procedures. Because they are highly specialized techniques, assistance in carrying them out should be sought from the Army Environmental Hygiene Agency.

5.2.1 Bioassay Programs

Bioassay programs should be established whenever there is a potential for internal contamination. Bioassays are appropriate for five purposes: 1) preparatory evaluation, 2) exposure control, 3) diagnostic evaluation, 4) removal of work restrictions, and 5) termination evaluation (ANSI N343-1978).

A. Preparatory Evaluation. Bioassays should be performed before an individual begins work that could result in an internal exposure. These evaluations are performed to determine the nature and extent of any prior exposure that could affect an individual's availability for job assignments. Knowledge of prior exposures is also helpful in distinguishing, in later bioassays, which exposures are not attributable to the present working environment.

B. Exposure Control. Bioassays should be performed periodically to ensure the adequacy of physical containment and contamination control measures. Personnel should be evaluated often enough so that unfavorable exposure trends can be identified. Bioassays may be required more frequently whenever new processes, procedures, controls, or equipment are put into use, to verify that protective measures are adequate. An increased frequency is also required whenever surface or air contamination is detected.

C. Diagnostic Evaluation. Bioassays are used after a known intake of radioactive material to determine the location and amount of the deposition; to provide data necessary for estimating internal dose rates, the fraction of the deposition retained in the body, and dose commitments; and to determine the necessity of work restrictions or referrals for therapy.

D. Removal of Work Restrictions. If an individual's internal dose rate has approached or exceeded applicable limits and the individual's work in radiation areas has been restricted, bioassays should be performed to determine whether the dose rate has decreased enough so that the work restrictions can be lifted.

E. Termination of Employment. Bioassays should be performed as a regular part of the formal termination sequence in order to determine the level of internal exposure attributable to the individual's job function.

5.2.2 Actions To Be Taken Upon Detection of an Intake

If a routine bioassay performed to assess control indicates an abnormal (i.e., unexpected) presence of a radionuclide in the body or excreta, further evaluations should be made to confirm that an intake has actually occurred. (False indication of an intake may result from contaminated skin in the case of in-vivo counting, or from contaminated samples in the case of radioanalysis of excreta.) The individual should be surveyed for external contamination, procedures for external decontamination should be used (see Chapter 7), and then another in-vivo measurement should be made. If the measured activity decreases, the contamination is probably external. Continue decontamination procedures until two consecutive measurements result in no significant change. If the measured activity remains constant and an intake cannot be ruled out, then radioanalysis of excreta should follow.

The interpretation of in-vivo counting data is influenced by a number of variables. Examples of equations that can be used to calculate internal dose are provided in Section 5.3. However, the interpretation of bioassay data requires trained personnel. The RPO should contact the Army Environmental Hygiene Agency for assistance. The radioanalysis of excreta and other body samples is also performed by the Army Environmental Hygiene Agency. If activity is found in excreta samples, the agency can provide assistance in

interpreting the data in light of Publications 10 and 10A (1968, 1971) of the International Commission on Radiological Protection (ICRP).

The cause of a confirmed intake should be investigated, especially if the contamination occurs in several persons or recurs from time to time in one person. The dose reduction methods discussed in Report No. 65 of the National Council on Radiation Protection and Measurements (NCRP 1980) should be considered for use under the supervision of medical personnel.

Section 5.3 INTERNAL DOSIMETRY CALCULATIONS

Dosimetry is the measurement of the radiation absorbed by an object. Calculations of internal dosimetry, or the radiation absorbed by the body's organs and tissues, serve two purposes: 1) to determine the amount of radioactive material that can be inhaled in air or ingested in water by an individual without a radiation dose limit being exceeded; and 2) to estimate the radiation dose an individual will receive from radioactive material that has already entered the body. In the first case, the calculations are used for preventive purposes, to limit the dose that might be received by setting limits for the uptake of radioactive material; in the second case, the calculations are used for diagnostic purposes, to determine the dose that will actually be received. The two uses of internal dosimetry calculations will be discussed separately.

5.3.1 Calculation of Acceptable Intake

Most federal regulations concerning safe concentrations of radionuclides in air or water are based on the recommendations of the ICRP in its Publication 2 (1959), Report of Committee II on Permissible Dose for Internal Radiation. However, ICRP has recently issued revised recommendations in ICRP Publication 30, and these recommendations are being considered for incorporation into the Environmental Protection Agency's "Federal Radiation Protection Guidance for Occupational Exposures" (Federal Register, January 23, 1981).

The major difference between the two ICRP publications lies in the sophistication of the dose calculations used. In ICRP 30, mathematical descriptions

of organ shapes are used, whereas in ICRP 2, organs of a rather nebulous shape are assumed. The limits in ICRP 30 also account for the radiation dose to an organ from radioactive material situated in an unrelated organ, and ICRP 30 uses a more complex model of radionuclide distribution kinetics (that is, the rate of radiation's absorption into the body, distribution within the body, and eventual excretion from the body) than does ICRP 2.

Because current regulations are based on the earlier ICRP publication, the material in the text of this section relates to ICRP 2. The terms used in ICRP 30 and the equations developed there for calculating the radiation dose to various body organs are discussed in Appendix A.

The ICRP 2 methodology for calculating acceptable intakes of radionuclides in air or water involves three steps:

1. determining the critical organ; that is, determining which organ or tissue of the body would be most damaged by a given radionuclide entering the body
2. calculating the maximum permissible body burden; that is, calculating the maximum amount of the radionuclide that can enter the body without the maximum acceptable dose limit for the critical organ being exceeded
3. calculating maximum permissible concentrations; that is, calculating how much of the radionuclide can be in air that is breathed or water that is drunk without the maximum permissible body burden being exceeded.

These steps are explained below.

A. Determining the Critical Organ. The critical organ or critical tissue is the organ or tissue that, if damaged by radioactive material taken into the body, would cause the greatest physiological damage to the body. In concept, the critical organ or tissue for a given radionuclide is determined by considering: 1) which organ accumulates the greatest concentration of the radionuclide; 2) the importance of each organ to the well-being of the entire body; 3) which organs are most affected by the route of entry of the radionuclide into the body (e.g., the lungs are most affected by the inhalation of a radionuclide); and 4) the radiosensitivity of each organ, that is, which organ is damaged by the lowest dose. In practice, the first criterion (the organ that

has the greatest concentration of a given radionuclide) is used in ICRP 2 to determine the critical organ because of the difficulty of evaluating the other criteria. If the radionuclide is not concentrated in any single organ, then the whole body is considered to be the critical organ.

B. Calculating the Maximum Permissible Body Burden. The maximum permissible body burden (MPBB) is the amount of a radionuclide, accumulated throughout the body of an individual over 50 years of occupational exposure, that will result in a maximum permissible dose-equivalent rate to the critical organ for that radionuclide. (See Chapter 3, Table 3.2, for maximum permissible dose-equivalent rates.)

C. Calculating the Maximum Permissible Concentrations in Air and Water. The MPBB must be considered in order to estimate the acceptable concentrations of a radionuclide in air or water. In ICRP 2, a maximum permissible concentration for air, $(MPC)_a$, and a maximum permissible concentration for water, $(MPC)_w$, are given. The $(MPC)_a$ and $(MPC)_w$ are calculated based on a constant intake of a radionuclide into the body and an exponential elimination of the radionuclide from the body by radioactive decay and biological excretion. The calculations account for the breathing rate of the individual in the case of $(MPC)_a$ and for the amount of water the individual might consume during the day in the case of $(MPC)_w$. The fraction of the material actually retained in the body is also considered. The ICRP 2 recommendations for $(MPC)_a$ and $(MPC)_w$ limits have been incorporated into the permissible concentrations of radionuclides in air and water that are listed in 10 CFR 20, Appendix B. The MPC is given in μCi/ml.

5.3.2 Estimation of Internal Dose

Following the ingestion or inhalation of radioactive material, three dose computations can be made: 1) the initial dose-equivalent rate, which is important because it serves as the basis for calculating the total dose received; 2) the dose equivalent the critical organ or the total body will receive over 1 year; and 3) the total dose equivalent the critical organ or the total body will receive as a result of the ingestion. The total dose equivalent can be calculated either for an infinite time following the ingestion or for 50 years following the ingestion. A calculation based on the 50-year period results in

what is called the 50-year dose commitment. The methods of dose calculation described in this section are for an individual of standard size and average metabolism (which affects the rate of excretion of the radioactive material). If the calculations are to be modified to fit a particular individual, the Army Environmental Hygiene Agency should be contacted for assistance.

Equations provided in ANSI Standard N343-1978 can be used for calculating the initial dose-equivalent rate and the 1-year and 50-year dose equivalents resulting from an intake of radioactive material. In all cases, it is necessary to know the amount of radioactive material in the body or in the organ for which the dose is being calculated. The calculations would be based on a single in-vivo measurement (i.e., a measurement of the radiation emitted from the body, made using an external detector soon after the intake).

A. Calculating the Initial Dose-Equivalent Rate to an Organ.

$$\dot{H} = \frac{51.2 \times q(t) \times f_2 \times \varepsilon}{m} \qquad (5.1)$$

where $\dot{H}$ = the dose-equivalent rate to the organ (rem/day)

$q(t)$ = the activity in the whole body at the time of measurement (μCi)[(a)]

f_2 = the fraction of the total-body radioactivity in the organ of reference, from ICRP 2[(a)]

ε = the effective absorbed energy per disintegration (MeV/dis)

m = the mass of the organ of reference (g)

51.2 = constant ([rem·g·dis]/[μCi·MeV·day]).

B. Calculating the One-Year Dose Commitment Based on a Single In-Vivo Measurement. Equation (5.1) allows the calculation of the dose-equivalent rate to an organ containing radioactive material. One may be more interested in the total dose an individual will receive for a year and/or a lifetime following a deposition. Equation (5.2) allows for the calculation of the 1-year dose equivalent to an organ containing radioactive material.

(a) If the amount of radioactive material actually in the organ of interest is known, then that activity, in units of microcuries, may be used in the equation rather than the product $f_2\ q(t)$.

$$H_y = \frac{51.2 \times q(t) \times f_2 \times \varepsilon \times e^{\lambda t} \ [1 - e^{-365\lambda}]}{m \times \lambda} \qquad (5.2)$$

where H_y = the 1-year dose equivalent based on a single in-vivo measurement (rem)

$q(t)$ = the activity in the whole body at the time of measurement (μCi)

f_2 = the fraction of the total-body radioactivity in the organ of reference

ε = the effective absorbed energy per disintegration (MeV/dis)

e = the base of the natural logarithms ($e = 2.71828$)

λ = the effective removal constant ($\lambda = 0.693/t_{eff}$) ($days^{-1}$)

t = the time between the intake and the in-vivo measurement (days)

m = the mass of the organ (g)

51.2 = constant ([rem·g·dis]/[μCi·MeV·day]).

C. Calculating the Fifty-Year Dose Commitment. The 50-year dose equivalent can be calculated by modifying the exponent (-365λ) in the above equation to (-18250λ), which corresponds to a 50-year time interval.

Values of f_2, λ, and ε for a few selected radionuclides are given in Table 5.1. The parameters f_2 and λ listed in this table are based on a "standard man," defined in the Radiological Health Handbook (1970) as having a body weight of 70 kg. The use of these values in an equation will provide an estimate of the radiation dose to an individual who is the same size as the standard man. If possible, bioassay procedures should be used to obtain estimates of f_2 and λ that more closely match the individual.

Another source of reference for calculating the 50-year dose commitment is NUREG-0172 (NRC 1977). This report lists 50-year committed radiation doses to selected organs following the chronic intake of several radionuclides over a 1-year period. The radiation doses are calculated in terms of mrem per 50 years per pCi (10^{-12} Ci) of radioactive material. The dose calculations are for populations rather than occupationally exposed individuals and include 1) radiation doses from liquid effluents, 2) radiation doses from gaseous

TABLE 5.1. Parameters for Internal Dosimetry[a]

Organ Mass (grams) for Standard Man

Total Body	70,000	Liver	1,700
Lung	1,000	Spleen	150
Thyroid	20	Bone	7,000

Nuclide	Organ	f_2	λ[b]	ε[c]	[1-exp(-365λ)]
^{3}H	Total Body	1.0	3.2 (-4)	0.01	0.11
^{54}Mn	Lung	(d)	8.1 (-3)	0.23	0.95
	Liver	1.0	3.0 (-2)	0.23	1.0
^{59}Fe	Lung	(d)	2.1 (-2)	0.42	1.0
	Spleen	0.02	1.7 (-2)	0.34	1.0
^{58}Co	Lung	(d)	1.5 (-2)	0.29	1.0
	Total Body	1.0	8.3 (-2)	0.61	1.0
^{60}Co	Lung	(d)	6.1 (-3)	0.72	0.89
	Total Body	1.0	7.3 (-2)	1.5	1.0
^{95}Zr-Nb	Lung	(d)	1.7 (-2)	0.52	1.0
	Total Body	1.0	1.2 (-2)	1.1	0.99
^{95}Nb	Lung	(d)	2.6 (-2)	0.26	1.0
	Total Body	1.0	2.1 (-2)	0.51	1.0
^{106}Ru-Rh	Lung	(d)	7.7 (-3)	1.4	
	Kidney	0.07	2.8 (-1)	1.3	1.0
	Total Body	1.0	9.6 (-2)	1.4	1.0
^{131}I	Thyroid	0.2	9.6 (-2)	0.23	1.0
^{133}I	Thyroid	0.2	8.0 (-1)	0.54	1.0
^{134}Cs	Lung	(d)	9.5 (-3)	0.57	0.97
	Total Body	1.0	1.1 (-2)	1.1	0.98
^{137}Cs-^{137m}Ba	Lung	(d)	5.8 (-3)	0.41	0.88
	Total Body	1.0	9.9 (-3)	0.59	0.97
^{140}Ba-La	Lung	(d)	6.0 (-2)	1.4	1.0
	Bone	0.7	6.5 (-2)	4.2	1.0
	Total Body	1.0	6.5 (-2)	2.3	1.0
^{144}Ce-Pr	Lung	(d)	8.2 (-3)	1.3	0.95
	Bone	0.38	2.9 (-3)	6.3	0.65
	Liver	0.19	4.7 (-3)	1.3	0.82
	Total Body	1.0	3.6 (-3)	1.3	0.73

(a) American National Standards Institute 1978.
(b) Units of day^{-1}.
(c) Units of (MeV/disintegration) x (rem/rad).
(d) Estimates of lung dose should be based on a measured lung burden. However, a total-body in-vivo measurement can be used to estimate an upper limit of the lung dose commitment by setting f_2 = 1.0 for the lung.

effluents, and 3) radiation doses from contaminated surfaces or volumes (i.e., external radiation).

The intake of the same amount of radioactivity can result in different radiation doses for people of different ages; consequently, four sets of dose factors are presented in NUREG-0172. The age groups considered are infant, child, teen, and adult. The 50-year dose commitment is calculated by reading the dose factor from the approriate table and multiplying this value by the number of picocuries taken into the body. The tables of NUREG-0172 are not reproduced in this manual.

REFERENCES

American National Standards Institute (ANSI). 1978. Internal Dosimetry for Mixed Fission and Activation Products. ANSI N343-1978. Washington, D.C.

American National Standards Institute (ANSI). 1980. Practices for Respiratory Protection. ANSI Z88.2, New York.

International Commission on Radiological Protection (ICRP). 1959. Report of Committee II on Permissible Dose for Internal Radiation. ICRP 2, Pergamon Press, Oxford.

International Commission on Radiological Protection (ICRP). 1968. Evaluation of Radiation Doses to Body Tissues From Internal Contamination Due to Occupational Exposure. ICRP 10, Pergamon Press, Oxford.

International Commission on Radiological Protection (ICRP). 1971. The Assessment of Internal Contamination Resulting from Recurrent or Prolonged Uptakes. ICRP 10A, Pergamon Press, Oxford.

International Commission on Radiological Protection (ICRP). 1978. Limits for Intakes of Radionuclides by Workers. ICRP 30, Part 1 and Supplement to Part 1, Pergamon Press, Oxford.

International Commission on Radiological Protection (ICRP). 1980. Limits for Intakes of Radionuclides by Workers. ICRP 30, Part 2, Pergamon Press, Oxford.

National Council on Radiation Protection and Measurements (NCRP). 1980. Management of Persons Accidentally Contaminated with Radionuclides. NCRP 65, Washington, D.C.

Radiological Health Handbook. 1970. U.S. Department of Health, Education, and Welfare, Washington, D.C.

U.S. Code of Federal Regulations. 1982. Title 10, Part 20, "Standards for Protection Against Radiation." U.S. Government Printing Office, Washington, D.C.

U.S. Department of the Army, Headquarters. Safety - Ionizing Radiation Protection (Licensing, Control, Transportation, Disposal, and Radiation Safety). AR 385-11, Washington, D.C.

U.S. Environmental Protection Agency (EPA). "Federal Radiation Protection Guidance for Occupational Exposures." Federal Register, January 23, 1981.

U.S. Nuclear Regulatory Commission (NRC). 1976. Manual of Respiratory Protection Against Airborne Radioactive Materials. NUREG-0041, Washington, D.C.

U.S. Nuclear Regulatory Commission (NRC). 1977. Age-Specific Radiation Dose Commitment Factors for a One Year Chronic Intake. NUREG-0172, Washington, D.C.

APPENDIX A

ICRP 30 RECOMMENDATIONS FOR LIMITING RADIONUCLIDE INTAKES

APPENDIX A

ICRP 30 RECOMMENDATIONS FOR LIMITING RADIONUCLIDE INTAKES

The most recent recommendations of the ICRP for safe limits of radioactive material in air and water are found in Publication 30, Limits for Intakes of Radionuclides by Workers. To date, ICRP 30 consists of two parts published in 1978 and 1980, each with a supplement. A third part and supplement are expected to be published. Because ICRP 30 is so recent, its recommendations have not been incorporated into current government regulations; however, they may be incorporated into future regulations.

A.1 EXPLANATION OF TERMS USED IN ICRP 30

The sequence of steps used in ICRP 30 to determine acceptable concentrations of radionuclides in air or water is identical to that used in ICRP 2 and discussed in Section 5.3. The terminology used in ICRP 30 is different from that used in ICRP 2, however, and is explained below.

A. Committed Dose Equivalent. In ICRP 30, the Commission is attempting to limit two types of radiation effects in the body: 1) stochastic effects are those that are increasingly likely to occur as the radiation dose increases (for example, genetic effects and malignant diseases such as cancer); 2) non-stochastic effects are those that are increasingly severe as the radiation dose increases and that are unlikely to occur at all below a certain threshold dose (for example, loss of hair, skin damage, and cataracts).

The incidence of stochastic effects is limited if the risk of such effects resulting from the radiation dose to any single organ or combination of organs in 1 year does not exceed the risk associated with a whole-body dose equivalent of 5 rem in any 1 year. The risk of stochastic effects is quantified by a weighting factor for each organ; the weighting factor is an attempt to scale both the relative importance of the organ to the well-being of the body, and the organ's relative radiosensitivity. The weighting factors can be used to obtain a dose equivalent, H_L, to a tissue that yields the same risk as 5 rem

to the whole body. The committed dose equivalent ($H_{T,50}$) in a tissue is the total radiation dose equivalent received by an organ or tissue during the 50 years following an intake. The maximum intake of a radionuclide is limited in ICRP 30 by the requirement that the sum of the ratios of $H_{T,50}/H_L$ in all irradiated tissues not exceed 1.0. It is not possible to directly compare the doses to the critical organs given in ICRP 2 with the annual doses to the critical organs given in ICRP 30 (see Table 5.2 below). This is because ICRP 30 restricts the sum of the doses received by all the tissues of the body, whereas ICRP 2 restricts the dose to the critical organ only.

TABLE 5.2. Weighting Factors Recommended in ICRP 30

Organ or Tissue	Weighting Factor	H_L (a) (rem)
Gonads	0.25	20
Breasts	0.15	33
Red bone marrow, lung	0.12	42
Thyroid, bone surfaces	0.03	167
Five other tissues receiving the greatest dose in the remainder of the body	0.30	83

(a) Dose equivalent to a tissue giving the same risk as 5 rem to the whole body.

In order to prevent nonstochastic effects, ICRP 30 limits the radiation dose equivalent to any organ over the 50 years following an intake (the committed dose equivalent) to 50 rem.

B. Annual Limit of Intake. In ICRP 30, the MPBB of ICRP 2 has been replaced by the annual limit of intake (ALI). The ALI is the amount of a radionuclide that can be ingested or inhaled such that the sum of the ratios $H_{T,50}/H_L$ in all the tissues irradiated is equal to 1. In addition, the committed dose equivalent to any organ cannot exceed 50 rem in 1 year.

The ALI is calculated based on a constant inhalation or ingestion over the year. Also considered is the rate at which radioactive material is eliminated from the body by both radioactive decay and excretion. The intake rate can be exceeded at times as long as the total yearly intake does not exceed the specified ALI.

C. Derived Air Concentration. The MPCs given in ICRP 2 have been replaced in ICRP 30 by a derived air concentration (DAC), which is the acceptable concentration of a radionuclide in air. The ICRP 30 recommendations are listed in units of Bq/m^3, which can be converted to μCi/ml by multiplying by the conversion factor 2.7×10^{-11} $(\mu Ci \cdot m^3)/(ml \cdot Bq)$. No derived water concentration is defined in ICRP 30, nor is any value given that would be equivalent to the MPCs. The only mention made of a maximum concentration allowable in air and water is that the total intake should be less than the ALI.

A.2 DEVELOPMENT OF EQUATIONS USED IN ICRP 30

A major change in ICRP 30 as compared to ICRP 2 is that the radiation dose to an organ is determined taking into account the radioactive material in other organs as well as in the organ of concern. This change is especially important for intakes of radionuclides that emit gamma rays, x rays, or neutrons by spontaneous fission.

The committed dose equivalent to an organ ($H_{T,50}$) is a product of the committed absorbed dose ($D_{T,50}$), the quality factor of the radiation (Q), and other modifying factors (N). For the time being, ICRP has stated that N is equal to 1. In the following paragraphs, the equations used in ICRP 30 for calculating the internal dose are developed.

A. Radiation Energy, E. The dose equivalent to an organ is related to, or proportional to (symbolized $\propto$), the energy of the radiation. In the case of alpha particles and gamma rays, E is the energy of the radiation listed on periodic tables and in reference books. In the case of beta particles, an average energy of the radiation must be calculated because beta particles are emitted from the nucleus with a spectrum of energies. As a general rule, the

average beta energy is about one-third of the listed or maximum energy. A more exact equation is:

$$E = 0.33 \times \left(1 - \frac{\sqrt{Z}}{50}\right) \times \left(1 + \frac{\sqrt{E_{max}}}{4}\right) \times E_{max} \qquad (5.3)$$

where E = average beta energy (MeV)
Z = atomic number of the emitting nucleus
E_{max} = maximum beta energy (MeV)
0.33 = constant.

For positrons, the equation is:

$$E = 0.33 \times \left(1 + \frac{\sqrt{E_{max}}}{4}\right) \times E_{max} \qquad (5.4)$$

where E = average positron energy (MeV)
E_{max} = maximum positron energy (MeV)
0.33 = constant.

The relation of the committed dose equivalent to the radiation energy is:

$$H_{T,50} \propto E \qquad (5.5)$$

where $H_{T,50}$ = committed dose equivalent to a target organ
E = energy of the radiation (MeV).

B. Type of Radiation Emitted. Each type of radiation has a characteristic rate of energy deposition, or linear energy transfer (LET), as described in Chapter 1. The quality factor, Q, is a function of the radiation's LET and is included in the calculation of the dose equivalent.

The relation can now be written as follows:

$$H_{T,50} \propto E \times Q \qquad (5.6)$$

where $H_{T,50}$ = committed dose equivalent to a target organ
E = energy of the radiation (MeV)
Q = quality factor of the radiation.

C. Mass of the Organ, m_T. The radiation dose received by an organ is inversely proportional to the mass of the organ. Because the absorbed dose is defined in terms of energy absorbed per unit mass, if the amount of energy deposited remains constant, then the absorbed dose necessarily decreases as the mass of the organ increases.

The relation for the committed dose equivalent is therefore:

$$H_{T,50} \propto \frac{E \times Q}{m_T} \tag{5.7}$$

where $H_{T,50}$ = committed dose equivalent to a target organ
E = energy of the radiation (MeV)
Q = quality factor of the radiation
m_T = mass of the target organ (g).

D. Absorbed Fraction of the Emitted Energy, AF(T←S). A fraction of the energy emitted by radioactive material is absorbed in the organ containing the material, and the remainder escapes. The energy that escapes from the organ may penentrate through the body and produce a radiation dose in another organ, or it may escape from the body. The fraction of the emitted energy absorbed in a given organ is symbolized by AF(T←S); T represents the target organ (the organ receiving the dose), and S represents the source organ (the organ containing the radioactive material). The target organ and the source organ may be the same organ, or they may be different organs of the body. As a result, it is now possible to calculate the radiation dose to an organ resulting from radioactive material in a different organ.

For the calculation of the absorbed fraction, radiations can be placed into two categories: nonpenetrating radiation and penetrating radiation.

Nonpenetrating radiation is radiation that loses all of its energy after traveling a short distance in tissue. Examples of nonpenetrating radiation are alpha particles, beta particles, and protons. If the organ containing the radioactive material is large compared to this distance, all the energy emitted is deposited in the organ containing the radioactive material. That is:

$$AF(T \leftarrow S) = \begin{cases} 0, \text{ if T is not S} \\ \\ 1, \text{ if T is S} \end{cases} \qquad (5.8)$$

Penetrating radiation is radiation that penetrates through the body, depositing energy both in the organ that contains the radioactive material and in other organs. Examples of penetrating radiation include x rays, gamma rays, and neutrons. The calculation of AF(T←S) for penetrating radiation is quite complex and virtually impossible without the aid of a computer. The computer is first programmed with a mathematical description of a man of average size, termed the reference man or standard man. This mathematical description is called a phantom and describes the shape, density, and relative locations of the various bones and organs of the body. The absorbed fraction is then calculated using a "Monte Carlo" computer calculation. A description of the basic principles behind these calculations follows. The "Monte Carlo" calculations, although equivalent to this description, are different in detail to save computer time.

The radioactive nuclei are assumed to be distributed uniformly throughout the source organ. A point within the source organ is picked. The computer model emits a photon of energy E in some direction picked at random from all possible directions. The photon is followed along its path; after it has traversed a very short distance, the probability of its interacting is calculated. The computer then "flips a coin" with this probability. If a "head" results from the coin flip, the photon is considered to interact at that point. If the interaction is Compton scattering (see Chapter 1), the angle is picked at random with a relative probability determined by the energy of the photon and by the interacting medium. The energy of a recoil electron for scattering at that angle is calculated and deposited at the interaction site. Similar procedures are followed for the photoelectric effect and pair production. The scattered photon is then followed in the same way. If a "tail" occurs on the first coin flip, the photon is allowed to travel another small distance and the probability of interaction is again calculated. This procedure is repeated until all the energy has been absorbed or the radiation leaves the

body. The entire procedure is repeated many times for each organ, until one has a map of the radiation deposited in all organs by gamma rays (or other penetrating radiation) leaving the specified point in the source organ. The result of these calculations is the AF(T←S) for penetrating radiation. These values are tabulated in ICRP 30 and its supplements.

The relation for the committed dose equivalent is now written as:

$$H_{T,50} \propto \frac{E \times Q \times [AF(T \leftarrow S)]}{m_T} \qquad (5.9)$$

where $H_{T,50}$ = committed dose equivalent to an organ
E = energy of the radiation (MeV)
$AF(T \leftarrow S)$ = absorbed fraction of the emitted energy
m_T = mass of the target organ (g)
Q = quality factor of the radiation.

E. Radiation Yield, Y. A radionuclide can undergo decay by different pathways. In the case of a beta-emitting nuclide, all pathways are similar in that they entail the emission of a beta particle followed by a gamma ray, but they differ from each other in the distribution of energy between the beta particle and the gamma ray. The radiation yield, Y, is the fraction of disintegrations that yield a certain radiation type and energy.

The committed dose-equivalent relation can now be written as:

$$H_{T,50} \propto \frac{Y \times E \times Q \times [AF(T \leftarrow S)]}{m_T} \qquad (5.10)$$

where $H_{T,50}$ = committed dose equivalent to an organ
Y = radiation yield (no units)
E = energy of the radiation (MeV)
$AF(T \leftarrow S)$ = absorbed fraction of the emitted energy
m_T = mass of the target organ (g)
Q = quality factor of the radiation.

The expression on the right side of Equation (5.10) is collectively referred to as the specific effective energy [SEE(T←S)]. This indicates the energy, in units of MeV, deposited per gram of the target organ for each

disintegration. Because the radioactive material may emit more than one type of radiation, it is necessary to sum the contributions from all radiations emitted; that is:

$$SEE(T \leftarrow S)_{total} = \sum_{i=1}^{n} [SEE(T \leftarrow S)]_i \qquad (5.11)$$

where

$SEE(T \leftarrow S)_{total}$ = specific effective energy of the nuclide (MeV/[g·Bq]), which is unique for any given combination of nuclide, source organ, and target organ

$$\sum_{i=1}^{n} [SEE(T \leftarrow S)]_i = [SEE(T \leftarrow S)]_{radiation\ 1} + [SEE(T \leftarrow S)]_{radiation\ 2} + \ldots + [SEE(T \leftarrow S)]_{radiation\ n}$$

Thus, we can write the relation for the committed dose equivalent as

$$H_{T,50} \propto SEE(T \leftarrow S) \qquad (5.12)$$

where $H_{T,50}$ = committed dose equivalent to an organ

$SEE(T \leftarrow S)$ = specific effective energy of the radioactive nuclide per disintegration (MeV/g·dis).

F. Total Number of Disintegrations in the Source Organ, U_S. The total number of disintegrations in an organ over the 50 years following a single uptake of radioactive material is a complicated function of the physical decay of the radionuclide and the metabolic characteristics of the chemical compound that contains the radionuclide. For example, radioactive material may be biologically eliminated from one organ, perhaps the lung, only to be absorbed by a second organ, such as the liver. The equations describing the time-dependent distribution of the radioactive material can be found in ICRP 30 and are not discussed here.

If the 50-year cumulated activity in the source organ, U_S, is given in disintegrations, then the relation for the committed dose equivalent may be written as:

$$H_{T,50} \propto U_S \times [SEE(T \leftarrow S)] \qquad (5.13)$$

where $H_{T,50}$ = committed dose equivalent to an organ
U_S = number of transformations in the source organ S over 50 years following the intake of a radionuclide.

G. Conversion Factors. Finally, the calculation of appropriate conversion factors allows the replacement of the proportionality symbol by an equal sign. The conversion factors convert the energy deposition to rem for the traditional system, or to sievert if the SI system is to be used. In units of rem, the appropriate equation is:

$$H_{T,50} = (1.6 \times 10^{-8}) \times U_S \times [SEE(T \leftarrow S)] \qquad (5.14)$$

where $H_{T,50}$ = committed dose equivalent to a target organ (rem)
U_S = number of transformations in the source organ S over 50 years following the intake of a radionuclide
$SEE(T \leftarrow S)$ = specific effective energy of the radionuclide (MeV/g).

In units of sievert, the equation is:

$$H_{T,50} = (1.6 \times 10^{-10}) \times U_S \times [SEE(T \leftarrow S)] \qquad (5.15)$$

where $H_{T,50}$ = committed dose equivalent to a target organ (Sv)
U_S = number of transformations in the source organ S over 50 years following the intake of a radionuclide
$SEE(T \leftarrow S)$ = specific effective energy of the radionuclide (MeV/g).

CHAPTER 6. EXTERNAL EXPOSURE

FIGURES

TABLES

CHAPTER 6. EXTERNAL EXPOSURE

External radiation exposure is the exposure of the body to radiation originating outside of the body. For example, an external radiation exposure may be received from radioactive material in a package, from fixed contamination on a bench top, or from an x-ray machine. The hazard presented by external radiation and the methods used to control external exposure are dependent upon the penetrating ability of the radiation and the dose rate encountered. Penetrating radiations such as photons and neutrons, which can pass into the body and irradiate the internal organs, are considered more hazardous than the relatively nonpenetrating charged particles, such as alpha and beta particles.

If a radioactive source material is shielded so that the radiation is emitted as a beam, then only those parts of the body that traverse the beam will be irradiated. This causes a partial-body irradiation. Common sources of severe partial-body irradiation are radiation-producing machines such as x-ray machines and accelerators, which are capable of producing intense beams of radiation. If the beam is large enough, or if the radioactive source material is not shielded, then the entire body may receive a dose of radiation; this is called a whole-body dose.

Exposure to external radiation can be controlled or reduced by a number of methods, primarily the judicious use of time, distance, and shielding. In this chapter, these and other methods are discussed, the monitoring of external doses is described briefly, and procedures for estimating external dose are given.

Section 6.1 CONTROL AND REDUCTION OF EXTERNAL RADIATION DOSE

The primary methods of reducing external radiation dose are the use of time, distance, and shielding. Other methods are also available. Each task involving radioactive material should be carefully evaluated to determine which control procedures are appropriate. The ALARA (as low as is reasonably achievable) philsophy should always be considered in the development of control procedures.

6.1.1 Exposure Time

The longer the time spent working in a radiation field, the higher the dose received. An individual's working time can be reduced if work is planned and if dry runs, complete in every detail except for the use of radioactivity, are performed before any work with radioactive materials or radiation-producing machines is begun.

A. Basic Principle. The total dose received at a given distance from a particular source is a linear function of the exposure time; that is, doubling the exposure time doubles the total dose, and halving the time halves the total dose. This relationship can be expressed by Equation (6.1):

$$D = \dot{D} \times t \qquad (6.1)$$

where D = radiation dose

$\dot{D}$ = radiation dose rate, or dose per unit time

t = time of exposure to radiation.

This equation assumes that the dose rate is constant during the exposure time.

Minimizing an individual's exposure time is one of the simplest ways of reducing the individual's total dose. For example, if the dose rate from an unshielded source is 2 rad/hr and the time of exposure is 30 minutes, then the radiation dose received is:

$$D = 2 \text{ rad/hr} \times 0.5 \text{ hr} = 1 \text{ rad}$$

However, if the time of exposure to the source can be reduced to 15 minutes, then the radiation dose received is:

$$D = 2 \text{ rad/hr} \times 0.25 \text{ hr} = 0.5 \text{ rad}$$

B. Control of Time. Time spent in a radiation area can be controlled by the use of timekeepers. This practice requires that the dose rate in a given work area be known. The maximum allowable residence time in the area can then be calculated using Equation (6.2):

$$t = \frac{D}{\dot{D}} \qquad (6.2)$$

where t = maximum allowable residence time in the radiation area

D = maximum dose to be received by the individual

$\dot{D}$ = dose rate of the source.

In instances of very high dose rates or where rigid control of exposure is needed, a timekeeper should be available for each individual. The timekeeper stands away from the radiation source but within sight of the individual. When the specified time has elapsed, the timekeeper notifies the individual, who then leaves the area. Personnel should be instructed to leave the area immediately and without question upon notification by the timekeeper.

C. Reduction of Time. Time spent working in a radiation area can be reduced by a number of methods; examples include training, the use of power equipment, easy access to equipment, and modification of the task to be performed.

The amount of time an individual spends in a radiation area can depend on how quickly and efficiently he or she can perform a task. Training can improve work efficiency and thus reduce exposure in day-to-day use of radioactive material.

Training programs should include actual performance of a procedure, complete in every detail (including the use of protective clothing, survey instruments, etc.) with the sole exception that radioactivity is absent. In some instances, this may mean that full-scale mockups constructed. Personnel can then practice the procedures, becoming more proficient and confident. At the same time, the procedures should be observed and analyzed by the Radiation Protection Officer (RPO) in an attempt to reduce the working time. Training is discussed in greater detail in Chapter 12.

The use of power equipment can reduce the time spent on a job. Examples of time-saving equipment include motorized carts for transporting materials in warehouses; impact wrenches; and power screwdrivers, saws, and drills. Most power tools can be used on the job without modification, although tools for specialized applications may require modifications. Equipment used in a radiation area should always be monitored for contamination before being removed from the area.

Efficient access to components, systems, or equipment can significantly reduce the time required for their operation, maintenance, repair, or replacement. The ease of access to equipment and components should be assessed when equipment or work areas are being designed and should be evaluated frequently in existing situations. For example, the fabrication of work platforms or the removal of obstructions may improve access to equipment and reduce the time spent in a radiation area.

Task modifications that result in decreased exposure time also reduce the radiation dose received. A conscientious review of all repetitious tasks is the best method of maintaining radiation exposure ALARA. After each task is completed, all participants should discuss the task and methods to improve performance. Task modifications may also be identified in training sessions. All standing operating procedures (SOPs) should be continually upgraded and improved.

6.1.2 Distance from the Source

Often, the time spent near a radiation source cannot be reduced. Personnel should then either work farther away from the radiation source or place shielding between themselves and the source.

A. Basic Principle. If time and shielding remain constant, then the radiation dose decreases as the square of the distance from the source of radiation. Consequently, the relationship between distance and dose rate is commonly called the inverse-square law. This relationship is illustrated in Figure 6.1.

The equation for the inverse-square law is:

$$\dot{D}_2 = \dot{D}_1 \times \frac{(s_1)^2}{(s_2)^2} \qquad (6.3)$$

where $\dot{D}_1$ = the dose rate at distance 1
$\dot{D}_2$ = the dose rate at distance 2
s_1 = distance 1
s_2 = distance 2.

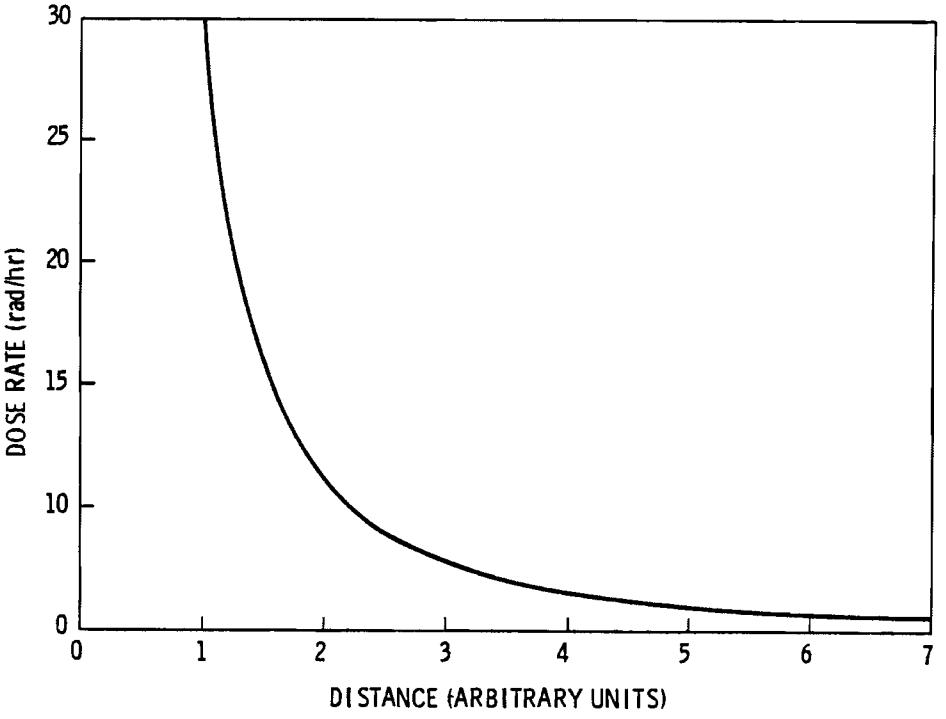


FIGURE 6.1. The Inverse-Square Relationship

The inverse-square law assumes that the radiation source is very small (a point source). If the distance between a nonpoint source and the irradiated object is at least five times the largest dimension of the source, then the inverse-square law can still be used. The inverse-square law also holds only in a vacuum. Attenuation of gamma rays and neutrons by air is usually negligible and does not influence the dose rate to an appreciable extent. However, alpha and beta particles are greatly attenuated by air, and as a result, inverse-square calculations overestimate the actual radiation dose for three types of radiation.

B. Control of Distance. Distance, as a method of reducing radiation exposure, can include remote operation, moving work away from radiation sources, and moving extraneous radiation sources away from the work area. Each task should be carefully evaluated to determine whether these procedures or others can be used to increase the distance between personnel and radiation sources.

(1) Remote Operation. Remote operation generally requires the use of a manipulating device, or remote-handling tool, to place distance between the operator and the radioactive source. For example, small radioactive sources are commonly equipped with a detachable handle or tool; most sealed sources come with a handling device; and forceps can be used to manipulate swipes for leak testing sources. Remote operations can also be performed using specially designed manipulators operated from behind barriers. Manipulators range in complexity from simple devices used in conjunction with temporary shielding to complex devices built into specially constructed hot cells.

Some manipulations are difficult to perform using remote-handling tools and can be performed faster and with a lower resultant dose using the fingers. However, direct handling of radioactive sources should be minimized and should be performed, when absolutely necessary, as quickly as possible to minimize the high dose rates that can result from direct handling. A 2-cm-diameter, 1-Ci source of ^{137}Cs, for example, gives a dose-equivalent rate of about 1.5×10^3 rem/hr to the hand when held in the hand. At this rate, the maximum allowable dose equivalent to the hand for one calendar quarter (18.75 rem) would be received in about 45 seconds. When sources must be handled directly, a finger dosimeter should be worn.

(2) Moving Away from Sources. A simple, often-overlooked technique for reducing exposure through the use of distance is for individuals to move away from the radiation source whenever possible. For example, if personnel need to discuss a procedure, they should move away from the source. If a defective part of a machine needs to be serviced, it should be removed and serviced elsewhere. Tradeoffs might be required if the object to be worked on is bolted onto or close to the radiation source and removal time exceeds servicing time. The ease of removing components should be considered during the design of equipment and of the building in which it is to be housed. Ideally, components that can be removed from the radiation area quickly and safely should be used.

Another example of moving away from the source is found in the use of gauging devices, such as those used to determine the surface density of road beds and the moisture content of roofs. During the operation of these devices,

the radiation source is moved from a well-shielded configuration to a less well shielded configuration. The operator should step back from the device while the timer is operating and the measurement is being taken.

(3) Removing Other Sources. Moving other sources away from the work area is the third method of using distance to reduce exposure. For example, piping can be backflushed to dislodge and remove radioactive debris. Other extraneous sources that should not be overlooked are contaminated stock bottles and accumulations of radioactive waste.

6.1.3 Shielding

Shielding is the use of barriers or absorbers placed between a source and an individual to stop some of the radiation reaching the individual. Alpha particles can be totally absorbed by a few centimeters of air or a few sheets of paper. Beta particles can be stopped by a few meters of air or a few millimeters of lead or plexiglass. Gamma radiation can penetrate even dense materials such as lead; however, the intensity of gamma radiation can be reduced to negligible levels by the use of shielding.

The attenuation of gamma radiation by an absorbing material can be described by the equation:

$$I = I_0 \times e^{-\mu s} \qquad (6.4)$$

where I = radiation intensity after traversing a thickness, s, of material

I_0 = original radiation intensity, i.e., the radiation intensity that would be observed had the attenuating material not been present

e = base of the natural logarithms ($e = 2.71828$)

μ = linear attenuation coefficient (cm^{-1})

s = thickness of the attenuating material (cm).

The linear attenuation coefficient, μ, is related to both the attenuating material and the energy of the photon. In many instances, the mass attenuation coefficient is available in references, rather than the linear attenuation coefficient.

The mass attenuation coefficient is the linear attenuation coefficient divided by the density of the medium. That is:

$$\mu_m = \mu/\rho \tag{6.5}$$

where μ_m = mass attenuation coefficient (cm^2/g)
μ = linear attenuation coefficient (cm^{-1})
ρ = density of the attenuating material (g/cm^3).

Thus

$$\mu = \mu_m \times \rho \tag{6.6}$$

Mass attenuation coefficients as a function of photon energies are listed for many materials in the Radiological Health Handbook (1970). The densities of common materials can also be found in the Radiological Health Handbook.

The half-value layer concept is useful in determining the necessary shielding for gamma radiation. A half-value layer (HVL) is the thickness of material required to reduce the radiation intensity by a factor of 2. This concept is similar to the half-life of radioactive decay. A related term, the tenth-value layer (TVL), is the thickness of an attenuating medium necessary to reduce the radiation intensity by a factor of 10. Both HVLs and TVLs for selected gamma sources and absorbing materials are given in Table 6.1.

TABLE 6.1. Half-Value and Tenth-Value Layers

Radionuclide	Half-Life	Gamma Energy (MeV)	Half-Value Layer (cm)			Tenth-Value Layer (cm)		
			Concrete	Steel	Lead	Concrete	Steel	Lead
^{60}Co	5.24 yr	1.17, 1.33	6.6	2.1	1.20	20.8	6.9	4.0
^{137}Cs	27 yr	0.66	4.8	1.6	0.65	15.7	5.3	2.1
^{192}Ir	74 d	0.13 to 1.06	4.3	1.3	0.60	14.7	4.3	2.0
^{198}Au	2.7 d	0.41	4.1	--	0.33	13.5	--	1.1
^{226}Ra	1622 yr	0.047 to 2.4	6.9	2.2	1.66	23.4	7.4	5.5

6.1.4 Other Methods of Controlling External Exposure

Inventory limitations, access restrictions, and a variety of other approaches can be used to help control exposure to external radiation.

A. Inventory Limitations. The hazard presented by radioactive material is a direct function of the quantity of material present. Inventories of radioactive material in laboratories can be reduced by the frequent collection of radioactive waste. An inventory of a radioactive chemical reagent can also be reduced by separating aliquots of the material into individual vials and storing the material that will not be used immediately away from the work area. The material can be separated by the user after receiving it, or it can be ordered in multiple containers from most suppliers, for a nominal fee. Two advantages result from this separation: 1) the radiation hazard resulting from spills or other accidents is reduced, and 2) inventory recordkeeping is simplified. The use of a centralized storage room for radioactive material not in use or used only occasionally is often convenient, relatively inexpensive, and secure. Such a facility is also helpful in keeping exposures ALARA, since less radioactive material is stored in laboratories or other areas occupied by personnel.

B. Good Practices. Other methods of reducing radiation exposures, which are discussed in more detail in other chapters of this manual, include the following:

1. Restrict access to areas that present a radiation hazard, through the use of locked doors, intrusion alarms, or guards. The means of restriction selected depends upon the radiation dose rates that are anticipated, the presence of interlocks, security restrictions, and budget.
2. Minimize the number of authorized radiation workers present by limiting the number of persons in an area at a given time.
3. Post signs in radiation areas. The work area should be surveyed every few months to ensure that the signs adequately describe the hazard associated with the area. The posting should indicate the actual hazard involved; do not "overpost." Habitual overstatement of radiation hazards may cause personnel to ignore the warning signs.

4. Keep copies of SOPs readily available to all radiation workers.

5. Maintain operating logs for all radiation-producing machines and radioactive sources. These logs should contain information such as date, time in, time out, and the names of the individuals working with the machines or sources. In some cases, it may be desirable to include the readings of a pencil dosimeter as each individual enters and leaves the area.

6. Use a "buddy system" so that an individual never works alone in a radiation area, particularly in one that is locked.

7. Establish areas that require an estimation of the dose rate before a person can enter.

Section 6.2 MONITORING OF EXTERNAL RADIATION DOSE

The primary DA dosimeter is the film badge (see Chapter 2). Pocket dosimeters and thermoluminescence dosimeters (TLDs) can be used to supplement the film badge. Supplementary dosimeters should be used when an individual is likely to receive more than 5 mrem in 1 hour and must be used when an individual enters a high-radiation area where the dose rate may be greater than 100 mrem/hr.

The dosimetry service for Army personnel and the responsibilities of the RPO in reviewing radiation doses to personnel are discussed in the following sections.

6.2.1 Dosimetry Service

Dosimeters for all personnel (army, civilian, and contractor) working with DA, ARNG, and USAR are provided by DARCOM. The dosimetry service is coordinated through the Lexington-BlueGrass Army Depot (Attn: AMXLX-ME-1), and an informational packet that describes the procedures for obtaining dosimetry services is available upon request. Because these procedures are updated periodically, they will not be detailed here. Actual requisitions for dosimetry service should be sent to the appropriate Army depot designated in the informational packet obtained from Lexington-BlueGrass.

When dosimetry service is requested for an individual, the RPO should be prepared to provide the following information about that person:

1. name of individual
2. date of birth
3. social security number
4. work classification
5. type of dosimeter required (i.e., whole-body or extremity)--If extremity, include the body part it is to be worn on (i.e., wrist, finger). If whole-body, include the radiation of interest (i.e., beta, gamma, x ray, or neutron). If a neutron badge is required, a beta-gamma badge should also be requested because neutron radiation is almost always accompanied by gamma radiation.

6.2.2 Review of Radiation Doses

The RPO is responsible for reviewing the radiation dose received by personnel (10 CFR 20, AR 40-14). These evaluations provide the basis for showing compliance with existing regulations and can be used to spot trends in doses received by personnel.

Dosimetry services that process dosimeters report personnel doses in terms of rem; no further calculations need to be performed by the RPO. The dose and the date the information is received are transferred onto each individual's record. The RPO should review the individual records at least once each calendar quarter to check for administrative overexposures and to spot any unusual trends in both individual and collective dose equivalents. If any trends are noted, especially increases in dose equivalents, an investigation should be conducted to determine the cause and correct any situations contributing to the increases. Criteria for judging whether an individual overexposure has occurred and for reporting any overexposures are discussed in Chapter 11, Section 11.3. Briefly, any monthly whole-body dose equivalent exceeding 500 mrem is categorized as an overexposure.

Section 6.3. ESTIMATION OF EXTERNAL RADIATION DOSE

Factors that affect the external radiation dose a person may receive from a radiation source include time, distance, shielding, and the activity of the source. The first three factors have already been discussed. The activity of the source material, often referred to as the source strength, has a direct linear relationship to the dose rate. That is, if the source activity is doubled, then the dose rate is doubled. Source activity is expressed as the activity of the parent radionuclide and is given in units of curies. Terms such as intense source, large source, or small source are relative terms and should be avoided.

Many methods can be used to estimate radiation doses from radioactive sources outside the body. The more sophisticated methods are computer-based calculations that must be performed by experienced individuals. However, for evaluating a facility's safety requirements, rapid estimates of radiation doses that are relatively accurate are often sufficient.

6.3.1 External Dose from Alpha Particles

An alpha particle must have an energy of at least 7.5 MeV to penetrate the 0.07-mm-thick protective layer of the skin. The vast majority of alpha-emitting radionuclides have alpha energies less than 7.5 MeV. For this reason, alpha particles do not present an appreciable external radiation hazard, and dose calculations are generally not required.

6.3.2 External Dose from Beta Particles

The dose rate 10 cm from a source of beta particles is given by Equation (6.7), which is valid over a wide range of beta energies.

$$\dot{D} = 2700 \times A \qquad (6.7)$$

where $\dot{D}$ = the dose rate (rad/hr)

A = the activity of the source (Ci).

In order to calculate the dose rate at distances other than 10 cm, the inverse-square relationship can be used. Equation (6.7) neglects the ability of air,

and even of the source material itself, to reduce or attenuate the dose rate. The attenuation of beta particles by air can be appreciable, and large errors in the calculated dose rate occur at distances beyond about 1 meter from the source.

The dose rate, in air, at the surface of a beta source is given by Equation (6.8):

$$\dot{D} = \frac{A}{S} \times P_i \qquad (6.8)$$

where $\dot{D}$ = dose rate (rad/hr)

A = source activity (mCi)

S = surface area of the source (cm^2)

P_i = specific ionization of the radiation, or the average number of ion pairs produced per centimeter of the radiation's path in air (taken from Table 6.2).

TABLE 6.2. Specific Ionization for Electrons[a]

Radiation Energy (MeV)	P_i (Ion Pairs/cm)	Range in Air (cm)
0.05	250	3.02
0.10	175	10.80
0.20	96	32.50
0.30	76	59.60
0.50	60	122.00
1.00	53	310.00
1.50	47	526.00

(a) Brodsky and Beard 1960.

6.3.3 External Dose from Gamma Radiation

Most equations for calculating the gamma-ray dose result in the exposure (the measure of the ionization of air by gamma radiation, measured in roentgen (R)), rather than the absorbed dose (rad) or dose equivalent (rem). The factors

for converting from exposure in units of roentgen to dose equivalent in units of rem are nearly equal to 1 for photons with energies greater than about 600 keV. Photons with energies less than about 600 keV are greatly scattered, resulting in a dose-equivalent rate in rem that is higher than the exposure rate in roentgen. Therefore, for photons with energies above 662 keV, the conversion factor 1.03 should be used, and for photons with energies below 662 keV, the conversion factors listed in Table 6.3 should be used. The three depths included in the table are for dose equivalents to 1) the whole body (1.0-cm depth, or deep dose equivalent); 2) the lens of the eye (0.3-cm depth); and 3) the skin (0.007-cm depth, or shallow dose equivalent).

TABLE 6.3. Conversion Factors for Computing Dose Equivalent from Exposure[(a)]

Photon Energy (keV)	Conversion Factor at a Depth of		
	1.0 cm ("deep")	0.3 cm	0.007 cm ("shallow")
15	0.28	0.67	0.90
20	0.58	0.79	0.94
30	1.00	1.07	1.11
40	1.28	1.29	1.34
50	1.46	1.46	1.50
60	1.47	1.47	1.52
70	1.45	1.45	1.50
80	1.43	1.43	1.48
90	1.41	1.41	1.45
100	1.39	1.39	1.43
110	1.37	1.37	1.40
120	1.35	1.35	1.36
130	1.33	1.33	1.34
140	1.32	1.32	1.32
150	1.30	1.30	1.30
662	1.03	1.03	1.03

(a) American National Standards Institute (ANSI) Standard N13.11-1978.

A. Exposure Rate from Any Gamma Point Source. A point source is a small source of radiation. The commonly used equations for calculating the exposure rate to an individual from a point source assume that the distance between the source and the individual is at least five times the diameter of the source or the diameter of the individual, whichever is larger. The simplest equation used to calculate the exposure rate from a gamma-emitting radionuclide is based on the specific gamma-ray constant (Γ) of the radionuclide, as given in Table 6.4.

$$\dot{X} = \frac{A\,\Gamma}{s^2} \tag{6.9}$$

where $\dot{X}$ = exposure rate (R/hr)
A = source activity (mCi)
Γ = specific gamma-ray constant ($[R \cdot cm^2]/[hr \cdot mCi]$)
s = distance from the source (cm).

If the specific gamma-ray constant for a gamma-emitting radionuclide is not listed in Table 6.4, then the following two equations can be used. For a distance from a source measured in meters:

$$\dot{X} = \frac{0.54\ A \sum_{i=1}^{k} E_i\, n_i}{s^2} \tag{6.10}$$

where $\dot{X}$ = exposure rate (R/hr)
A = source activity (Ci)
E_i = energy of photon i (MeV)
n_i = number of photons of energy E_i emitted per disintegration
$\sum_{i=1}^{k} E_i\, n_i = E_1\, n_1 + E_2\, n_2 + \ldots\ldots E_k\, n_k$
s = distance from the source (m)
0.54 = constant ($[R \cdot m^2]/[MeV \cdot hr \cdot Ci)]$.

TABLE 6.4. Gamma Radiation Levels for One Curie of Some Radionuclides[a]

Nuclide	Γ[b]	Nuclide	Γ[b]	Nuclide	Γ[b]
Actinium-227	∿2.2[c]	Gold-198	2.3	Potassium-43	5.6
Antimony-122	2.4	Gold-199	∿0.9	Radium-226	8.25
Antimony-124	9.8	Hafnium-175	∿2.1	Radium-228	∿5.1
Antimony-125	∿2.7	Hafnium-181	∿3.1	Rhenium-186	∿0.2
Arsenic-72	10.1	Indium-114m	∿0.2	Rubidium-86	0.5
Arsenic-74	4.4	Iodine-124	7.2	Rutherium-106	1.7
Arsenic-76	2.4	Iodine-125	∿0.7	Scandium-46	10.9
Barium-131	∿3.0	Iodine-126	2.5	Scandium-47	0.56
Barium-133	∿2.4	Iodine-130	12.2	Selenium-75	2.0
Barium-140	12.4	Iodine-131	2.2	Silver-110m	14.3
Beryllium-7	∿0.3	Iodine-132	11.8	Silver-111	∿0.2
Bromine-82	14.6	Iridium-192	4.8	Sodium-22	12.0
Cadmium-115m	∿0.2	Iridium-194	1.5	Sodium-24	18.4
Calcium-47[d]	5.7	Iron-59	6.4	Strontium-85	3.0
Carbon-11[d]	5.9	Krypton-85	∿0.04	Tantalum-182[d]	6.8
Cerium-141	0.35	Lanthanum-149	11.3	Tellurium-121	3.3
Cerium-144	∿0.4	Lutecium-177	0.09	Tellurium-132	2.2
Cesium-134	8.7	Magnesium-28	15.7	Thulium-170	0.025
Cesium-137[d]	3.3	Manganese-52	18.6	Tin-113	∿1.7
Chlorine-38	8.8	Manganese-54	4.7	Tungsten-185	∿0.5
Chromium-51	1.16	Manganese-56	8.3	Tungsten-187	3.0
Cobalt-56	17.6	Mercury-197	∿0.4	Uranium-234	∿0.1
Cobalt-57	0.9	Mercury-203	1.3	Vanadium-48	15.6
Cobalt-58	5.5	Molybdenum-99	∿1.8	Xenon-133	0.1
Cobalt-60	13.2	Neodymium-147	0.8	Ytterbium-175	0.4
Copper-64	1.2	Nickel-65	∿3.1	Yttrium-88	14.1
Europium-152	5.8	Niobium-95	4.2	Yttrium-91	0.01
Europium-154	∿6.2	Osmium-191	∿0.6	Zinc-65	2.7
Europium-155	∿0.3	Palladium-109	0.03	Zirconium-95	4.1
Gallium-67	∿1.1	Platinum-197	∿0.5		
Gallium-72	11.6	Potassium-42	1.4		

(a) Radiological Health Handbook 1970.
(b) Γ = specific gamma-ray constant = $R \cdot cm^2/hr \cdot mCi$, or $\Gamma/10 = R \cdot m^2/hr \cdot Ci$.
(c) ∿ = approximately.
(d) A Manual of Radioactivity Procedures 1961, Appendix A, pp. 137-140.

When the distance from the source is measured in feet, an approximation of the exposure rate is given by Equation (6.11).

$$\dot{X} = \frac{6\ A \sum_{i=1}^{k} E_i\ n_i}{s^2} \qquad (6.11)$$

where $\dot{X}$ = exposure rate (R/hr)

A = source activity (Ci)

E_i = energy of photon i (MeV)

n_i = number of photons of energy E_i emitted per disintegration

s = distance from the source (ft)

6 = constant ($[R \cdot ft^2]/[MeV \cdot hr \cdot Ci]$).

B. Other Methods of Calculating Gamma Exposure. In special cases, such as for calculating of gamma dose from line sources or from planar disc sources, more complex equations than those listed above are needed. These equations, presented in Appendix A, are for estimating exposure based on the intensity of the photon radiation.

REFERENCES

American National Standards Institute (ANSI). 1978. Criteria for Testing Personnel Dosimetry Performance. ANSI N13.11, New York.

Brodsky, A., and G. V. Beard. 1960. A Compendium of Information for Use in Controlling Radiation Emergencies. TID-8206 (Rev.), U.S. Atomic Energy Commission, Washington, D.C.

National Council on Radiation Protection and Measurements (NCRP). 1961. A Manual of Radioactivity Procedures. NCRP 28, Washington, D.C. Also published in 1961 as National Bureau of Standards Handbook No. 80, Washington, D.C.

Radiological Health Handbook. 1970. U.S. Department of Health, Education, and Welfare, Washington, D.C.

U.S. Code of Federal Regulations. 1982. Title 10, Part 20, "Standards for Protection Against Radiation." U.S. Government Printing Office, Washington, D.C.

U.S. Department of the Army and Defense Logistics Agency. Medical Services - Control and Recording Procedures for Exposure to Ionizing Radiation and Radioactive Materials. AR 40-14, DLAR 1000.28, Washington, D.C.

APPENDIX A

ESTIMATION OF EXTERNAL GAMMA DOSE

APPENDIX A

ESTIMATION OF EXTERNAL GAMMA DOSE

Equations were presented in Section 6.3.3 for estimating the exposure rate from point sources of gamma radiation. A slightly more complicated method of dose estimation involves first calculating the flux, or intensity, of the radiation, which is measured in photons per unit area in unit time (usually in photons/cm^2·sec), and then using the flux to calculate the absorption of the radiation's energy by body tissues.

To calculate the flux from any source, it is necessary to consult a decay scheme to determine the number of photons emitted per disintegration. Cobalt-60, for example, emits two gamma rays per disintegration, and both of these must be taken into account in the calculation of the flux.

A.1 FLUX FROM A POINT SOURCE

For a point source, the photon flux can be calculated from:

$$I = \frac{(3.7 \times 10^{10}) \times A \times n}{4 \times \pi \times s^2} \qquad (6.12)$$

where
- I = photon flux for photons of a given energy (photons/[cm^2·sec])
- A = source activity (Ci)
- n = fraction of disintegrations that yield a gamma ray of a given energy (photons/disintegration)
- s = distance from the source (cm)
- π = pi = 3.1416
- 3.7×10^{10} = constant (disintegrations/[sec·Ci]).

A.2 FLUX FROM A LINE SOURCE

A typical problem might entail calculating the dose rate from a pipe that contains radioactive material. In principle, the problem could be solved by

considering the pipe (or line) to be a series of point sources, calculating the flux from each point, and then adding up the dose rates from all the points. At best, this would be tedious. Therefore, the following equation has been derived to calculate the photon flux from a line source. The equation is valid for any point, p, along the source.

$$I = \frac{(3.7 \times 10^{10}) \times A_s \times n}{4 \times \pi \times s} \times (|\theta_1| + |\theta_2|) \qquad (6.13)$$

where I = photon flux for photons of a given energy (photons/[cm^2·sec])

A_s = source activity per unit length of pipe (Ci/cm)

n = fraction of disintegrations that yield a gamma ray of a given energy (photons/disintegration)

π = 3.1416

s = distance from the pipe (cm)

θ_1, θ_2 = the angles shown in Figure 6.2 (radians)

3.7×10^{10} = constant (disintegrations/[sec·Ci]).

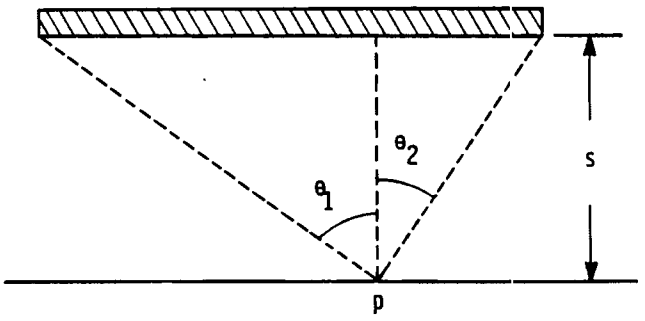


FIGURE 6.2. Line Source

A. FLUX FROM A PLANE DISK SOURCE

The dose rate from a plane disk source can be used to approximate the dose received from radioactive material on the ground. The photon flux at a point, d, from a plane disk source can be estimated from the equation:

$$I = \frac{(3.7 \times 10^{10}) \times A_s \times n}{4} \times \log\left[\frac{R^2}{s^2} + 1\right] \qquad (6.14)$$

where I = photon flux for photons of a given energy (photons/[cm^2·sec])

A_s = source activity per unit area (Ci/cm^2)

n = fraction of disintegrations that yield a photon of a given energy (photons/disintegration)

R = radius of the source (cm)

s = distance from the source (cm), as shown in Figure 6.3

3.7×10^{10} = constant (disintegrations/[sec·Ci]).

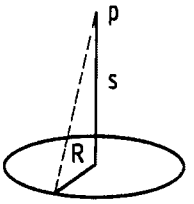


FIGURE 6.3. Plane Disk Source

A.3 ABSORPTION OF ENERGY BY TISSUES

The absorption of energy by body tissues is given by the energy absorption coefficient for the radiation in tissue. The basic equation is:

$$\dot{X} = 5.75 \times 10^{-5} \sum_{i=1}^{k} I_i \, (\mu_{en})_i \, E_i \qquad (6.15)$$

where $\dot{X}$ = exposure rate (R/hr)

I_i = photon flux (photons/[cm^2·sec])

$(\mu_{en})_i$ = mass energy absorption coefficient (cm^2/g)

E_i = the photon energy (MeV)

5.75×10^{-5} = constant ([R·g·sec]/[MeV·hr·photon])

$$\sum_{i=1}^{k} I_i \, (\mu_{en})_i \, E_i = I_1 \, (\mu_{en})_1 \, E_1 + I_2 \, (\mu_{en})_2 \, E_2 + \ldots + I_k \, (\mu_{en})_k \, E_k$$

The product of μ_{en} E for several photon energies is given in Table 6.5.

TABLE 6.5. Gamma-Ray Energy Absorption in Tissue[a]

Photon Energy (MeV)	μ_{en} E
0.2	0.0055
0.5	0.0164
1.0	0.0308
1.5	0.0422
2.0	0.0514
3.0	0.0675

(a) Radiological Health Handbook 1970.

CHAPTER 7. DECONTAMINATION

FIGURES

TABLES

CHAPTER 7. DECONTAMINATION

The presence of contamination, or unwanted radioactivity, can result from normal operations, maintenance activities, and abnormal events such as equipment failure, accidents involving radioactive materials, and improper work practices. Detecting and determining the extent of contamination usually require use of the survey techniques described in Chapter 4 of this manual. When the extent of contamination has been determined and appropriate barriers have been established to limit further spread, the process of cleanup, or decontamination, can begin.

Decontamination has three purposes: 1) to prevent any uptake of radioactive material into the human body; 2) to limit external radiation exposure; and 3) to prevent further spread of contamination. Decontamination may be required for personnel, for equipment of all types and sizes, and for large surface areas such as land, floors, roads, or buildings. The basic method of decontamination is to remove radioactivity by one or more wet or dry processes. Two other approaches that decrease the level of removable contamination are allowing short-lived radionuclides to dissipate through radioactive decay and fixing contamination in place by covering or sealing it. These approaches are not generally recognized as decontamination processes; however, under some circumstances they may be the best possible actions. For that reason they are included in this chapter.

Section 7.1 GENERAL DECONTAMINATION PROCEDURE

The specific decontamination methods and procedures selected for use in particular circumstances depend on the type, extent, and location of the contamination; however, the general approach to decontamination outlined below applies to most situations.

1. Control access to contaminated areas.
2. Provide personnel protection, including appropriate clothing, for workers.

3. Evaluate what is to be decontaminated.
4. Obtain necessary equipment and materials.
5. Survey all items to be released to an unrestricted area.
6. Begin with the mildest decontamination methods and progress to harsher, more abrasive, or caustic methods as required.
7. Work from the outside of the contaminated area to the inside.
8. Isolate all clean areas from contaminated areas. Clean areas adjacent to those being decontaminated should be covered with taped-down paper or plastic to prevent recontamination.
9. Minimize the generation of contaminated liquids and airborne radioactivity during the work, and collect and treat as contaminated waste all liquids generated and materials used during decontamination.
10. Survey between major steps in the decontamination process (i.e., between successive applications of each technique and between different techniques).
11. Continue decontamination until contamination levels are reduced to appropriate levels as given in Chapter 4, Appendix A, of this manual.
12. Document the completion of decontamination, including the name of the individual performing the final survey, the date, and the survey results. (Documentation of intermediate survey results may also be desirable.)

These steps are discussed further in the following sections on preparation for decontamination and on methods for decontaminating personnel, equipment, and materials. Specific procedures for applying these methods are given in the appendixes at the end of this chapter.

Section 7.2. PREPARATION FOR DECONTAMINATION

Preparation for decontamination includes establishing boundaries within which contamination is to be contained and controlling access to the area;

providing radiation protection for personnel involved in the decontamination operation; and evaluating the specific items to be decontaminated.

7.2.1 Area Definition and Access Control

Contaminated areas (e.g., floor or land areas) should be posted and barriers established to limit access to and further spread of contamination. In more complex situations (e.g., pieces of contaminated equipment or several rooms within a building), it may be necessary to segregate and isolate areas of relatively high contamination from those of relatively low contamination. Segregation can be useful in determining what effort will be required to complete decontamination, and it helps in the establishment of priorities, or a sequence for the work.

7.2.2 Personnel Protection During Decontamination

Radiation protection requirements for decontamination operations are the same as those for work in contaminated or high-dose-rate areas. The key concerns are to protect personnel from becoming contaminated and to keep both individual and collective radiation doses at levels that are as low as is reasonably achievable (ALARA).

Personnel can be protected against contamination by the use of protective clothing. For decontamination operations involving tritium, organic solvents, or other wet substances, clothing impervious to the liquids involved should be selected to prevent absorption of contamination through the skin. Respiratory protection should be used in highly contaminated areas, particularly when decontamination methods may generate or stir up loose contamination. Step-off pads should be positioned at exits from the contaminated area.

The radiation dose to personnel during decontamination can be monitored and controlled using standard instruments and techniques (e.g., thermoluminescence dosimeters, pocket dosimeters, dose rate monitoring, and surveys of individuals).

7.2.3 Evaluation of Decontamination Needs

Many materials such as wood, damaged equipment, scrap metal, cables, cords, hoses, and clothing require more time and effort to decontaminate than

they are worth. In general, these items should be disposed of as contaminated waste (see Chapter 10). If proper control procedures are used, some contaminated items can be assigned for use in permanently contaminated areas (e.g., in nuclear reactor facilities or radiochemistry laboratories).

Decontamination is begun at the perimeter of a large contaminated area and progresses toward the center. When appropriate, decontamination is from top to bottom of vertical surfaces. Perimeters should be surveyed and reestablished as the size of the contaminated area is reduced. The environment or topography may impose additional considerations for sequence; on sloping or windy terrain, decontamination should begin with the highest or upwind points, respectively. The presence of drains, sumps, or sewers warrants special consideration. Where they exist specifically for the collection of radioactive liquids, they should be used during decontamination; however, if they could become pathways for the further spread of contamination to the environment, every effort should be made to ensure their isolation.

Where areas with varying degrees of contamination can be identified, adequately segregated, and controlled, the priority for decontamination is less critical. In general, work should begin where the most significant reduction in personnel dose can be achieved through early decontamination. Other factors that may contribute to setting decontamination priorities include the availability of materials, equipment, and personnel, and how immediate the need is for uncontrolled access to or use of the area or equipment to be decontaminated.

Section 7.3. PERSONNEL DECONTAMINATION

Before external decontamination of an individual is begun, the following steps should be taken to help establish priorities for decontamination and for follow-up efforts:

1. Observe any physical effects to the contaminated person, such as bleeding, irregular breathing rate, burns, or shock.
2. Assess the extent of any injuries: medical treatment of injuries takes priority over decontamination.

3. Immediately flush with water any skin contamination involving caustic, corrosive, or organic-solvent solutions.

4. Determine the extent and magnitude of contamination using personnel survey techniques.

5. Document survey results.

6. Remove contaminated clothing, place it in a plastic bag, and hold it for further disposition.

7. Obtain assistance from medical personnel if decontamination of eyes, ears, nose, or mouth is necessary or if harsh chemicals (other than soap and water) will be required.

8. Investigate to determine how the contamination occurred.

For accident situations involving both contamination and personnel injury, medical treatment must take priority over decontamination. The only exceptions to this are 1) when an extremely high level of contamination presents a greater hazard to the victim than does the physical injury, and 2) when decontamination can be performed prior to treatment of minor injuries, and the medical officer concurs. In all cases, decontamination must be performed in a manner that prevents indiscriminate spreading of contamination.

When personnel contamination is suspected or detected, a thorough personal survey should be performed. Contaminated clothing should be removed and bagged for subsequent disposition. During the survey, particular attention should be paid to locating any hot spots of contamination. The results of this survey, including the locations and measured levels of contamination, should be documented. Figure 7.1 is an example of a data sheet for assessing personnel contamination. Refer to AR 385-40 to determine whether an accident/incident report is required.

In the event of a known or suspected internal deposition of radioactivity (by inhalation, injection, consumption, etc.), arrangements must be made for a prompt bioassay (see Chapter 5) and for consultation with the medical staff. The treatment and removal of internally deposited radioactivity is a highly specialized field, and the assistance of qualified medical personnel is essential.

PERSONNEL CONTAMINATION RECORD

Name:_____ Social Security Number:_____

Date of Incident:_____ Time of Occurrence:_____

Location of Incident:_____

_____

Description of How Contamination Occurred:_____

_____

_____

_____

_____

How was contamination discovered?_____

SURVEY RESULTS

Survey Performed by:_____

Survey Instrument Manufacturer and Model:_____

Serial Number:_____ Probe Type:_____

Indicate type, extent, and magnitude of contamination on figure below.

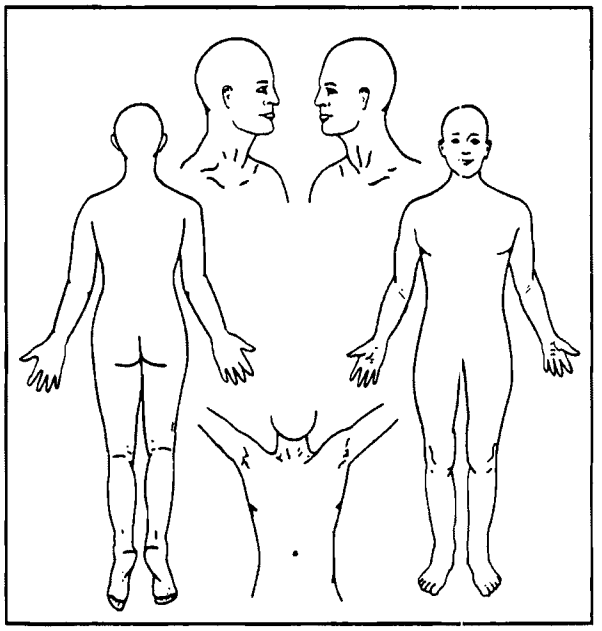


FIGURE 7.1. Personnel Contamination Record

7.3.1 Personnel Decontamination Methods

Personnel should be decontaminated as quickly as possible using the least drastic means necessary. Decontamination efforts should begin with mild methods, which should be continued as long as they are effective, and progress to harsher methods only as required. Medical supervision is required when harsh materials or methods are used. Extreme care should be taken to prevent the spread of contamination to any skin or body opening, and all liquids generated and materials used during decontamination should be collected and treated as contaminated waste. Personnel performing the decontamination should take all necessary precautions to protect themselves.

The progress of decontamination should be closely monitored by surveying between successive washings or techniques. A log of methods used and survey results should be maintained. A typical log sheet for personnel decontamination is shown in Figure 7.2.

Basic methods for personnel decontamination are listed in Table 7.1 in increasing order of harshness, along with their advantages, disadvantages, and decontaminating action, and some commonly available agents for each method. This is not a complete listing; many other agents have also been used effectively.

Simple washing methods (mild soaps, abrasive soaps, and detergents) are straightforward in their use. Generally, mild soap and water is sufficient for localized skin decontamination. A modification of simple washing is to make a paste by applying a powdered household laundry detergent to wet skin and rubbing. This method provides somewhat more effective decontamination, although it is also more irritating to the skin. Cool or lukewarm water should be used for all washing and rinsing. Hot water causes the skin pores to open, driving contamination deeper into the skin. Cold water closes the pores, trapping contamination in the skin.

If extensive washing is required or harsher methods must be used, obtain assistance from medical personnel before proceeding. In these circumstances, particular attention must be given to preventing skin damage. Chapping or cracking of the skin from repeated washing or abrasion can lead to the intake

DECONTAMINATION RECORD

Initial Contamination Level _____

Time Decontamination Begins	Skin Area Concerned	Decontamination Agent Used	Contamination Level After Decontamination	Skin Condition

Decontamination Completed by: _____ Time: _____

BIOASSAY (Check as applicable. Attach results when available.)

☐ in-vivo count ☐ nasal swipe

☐ urine sample ☐ feces sample

☐ none required ☐ other (specify) _____

FOLLOW-UP

Further evaluation needed? Yes_____ No_____

Type? _____

_____

Similar to previous occurrences? Yes_____ No_____

Explain _____

_____

Steps taken to prevent recurrence: _____

_____

_____

Comments: (attach if more space required)

_____

_____

_____

Radiation Protection Officer: _____ Date: _____

Reviewed by: _____ Date: _____

FIGURE 7.2. Decontamination and Evaluation Log

TABLE 7.1. Personnel Decontamination Methods

Method[a]	Common Agents	Action	Advantages	Disadvantages
Tape	Adhesive tape, masking tape	Removes by adhesion of contamination to tape	Simple, useful for spot contamination	Not useful for area contamination
Flushing	Water	Removes by flushing	Removes contamination if used immediately. May be used (with medical supervision) for eyes, ears, nose, mouth, and wounds	When used for nose and mouth, contaminated person should be warned not to swallow rinses
Mild soap and water	Bar soap, liquid soap	Emulsifies and dissolves contaminant	Readily available; effective for most cases	Continued washing defats skin. May spread contamination to other parts
Abrasive soap and water	Pumice-impregnated bar soap, powdered grit soap	Emulsifies, dissolves, and abrades	Readily available; effective for tough, calloused skin	Continued washing abrades skin
Detergent and water	Household laundry detergents	Same	Slightly more effective than soap	Will defat and abrade skin; use with care
Chemical complexing	10% EDTA solution	Chelates (bonds to contaminant)	Useful for heavy metals	
Oxidizing agents	Household bleach, potassium permanganate and sodium bisulfite	Dissolves contaminant absorbed in the epidermis	Superior for skin decontamination	Removes a layer of skin
Sweating[b]		Removes by sweating	Cleansing is from inside out	Contamination may seep into skin pores if profuse sweating is prolonged

(a) Listed in increasing order of harshness. Begin decontamination using mild methods and progress to harsher methods only as required. Medical supervision is required for all methods harsher than the use of soap and water.
(b) Sweating is a passive, mild decontamination method that should be used when other methods have been discontinued because of skin irritation or decreased effectiveness.

of radioactivity through minor cuts. The use of a hand cream or lotion between washings can help prevent chapping. If contamination still remains after extensive washing, covering the contamination with plastic (e.g., a plastic glove taped over the hand) and allowing the skin to sweat can provide further decontamination.

Chemical complexing agents, which should be used only under medical supervision, remove contamination by chemical interactions such as ion exchange and bonding. A solution of EDTA (ethylene-diamine-tetra-acetic acid) can be prepared by dissolving 10 grams of EDTA salts in 100 ml of water. This solution, which can be prepared in advance and stored, is applied to the skin with cotton swabs or sponges. Following each application of the solution, the area should be rinsed.

Oxidizing agents decontaminate by chemically removing the contaminant and a thin layer of skin. Household bleach is a weak oxidizing agent that can be applied full strength using cotton swabs or sponges. A stronger oxidizing agent is potassium permanganate ($KMnO_4$) followed by sodium bisulfite ($NaHSO_3$). Saturated solutions of each of these chemicals should be made up at the time of need by dissolving crystals of each in a small amount of water. (A saturated solution is one in which no more crystals will dissolve.) The $KMnO_4$ solution is painted thickly onto the skin and allowed to dry. It is then removed by gently scrubbing with the $NaHSO_3$ solution. The skin should be rinsed after each use of oxidizing agents, and their use should be discontinued if the skin becomes tender. Medical supervision is required for the use of this method.

Commercial decontamination agents--soaps, detergents, and complexing agents--are available under various trade names. They should be used with medical assistance and the manufacturer's instructions should be followed.

7.3.2 Specific Personnel Decontamination Procedures

Specific procedures for personnel decontamination are provided in Appendix A of this chapter. Procedures for decontaminating the skin, hair and scalp, body, face, eyes, ears, mouth, and nose are included.

7.3.3 Personnel Decontamination Kit

A personnel decontamination kit should be assembled for field use, or supplies should be available at designated decontamination stations. Typical materials that should be included are as follows:

Item	Approximate Quantity
Applicators, cotton-tipped	500
Cotton balls	200
Cleansing tissues	4 boxes
Sterile gauze pads (5 cm x 5 cm)	400
Hand brushes	4
Masking tape	1 roll
Plastic cups (4 oz.)	25
Plastic cups (1 oz.)	25
Plastic bags (for waste)	20
Scissors	1 pair
Surgical gloves (talced)	1 box
Flexible tube	1.2 meters
Filter paper (for smears)	1 box
Envelopes (to hold smears)	1 package
Hand cream	1 jar
Soaps: Regular bar soap	2 bars
Abrasive soap	1 bar
Detergent (household laundry type)	1 box
Reagents: Household bleach	1 bottle
Potassium permanganate crystals	1 small jar
Sodium bisulfite crystals	1 small jar
EDTA salts	1 small jar
Basin (for field use)	1
19-liter jug (for field collection of liquids)	1
Pencils or pens	3
Paper	1 pad

Section 7.4. EQUIPMENT AND MATERIAL DECONTAMINATION

Equipment and materials may need to be decontaminated for a number of reasons, including:

1. for release for unrestricted use
2. for the salvage of valuable material
3. to reduce the potential for exposure of personnel to radiation
4. to reduce the volume of contaminated waste.

Decontamination should be performed as soon as possible after contamination occurs. This is particularly true for liquid contaminants, which can penetrate farther into materials as contact time increases.

Materials that cannot be easily or cost-effectively decontaminated should be evaluated for possible limited use in restricted areas, or disposed of. Porous items (such as wood and unsealed concrete), intricately designed equipment, and items of low replacement cost tend to fall in this category.

7.4.1 Decontamination Methods

Many methods and techniques have been developed for decontaminating equipment (TM 3-220). Most are physical or chemical cleaning processes. Two other methods, which are not considered true decontamination, are radioactive decay (aging) and sealing contamination in place.

A. Cleaning, Abrasive, Chemical, and Electrochemical Methods. True decontamination entails removing radioactivity by cleaning, abrasive, chemical, and electrochemical methods. Cleaning methods are nondestructive but may require that equipment be disassembled for maximum effectiveness. Cleaning includes both manual (wiping, mopping, vacuuming) and mechanical (soaking, spraying, vibrating) techniques. Abrasive methods are destructive, involving the progressive removal of the contaminated material. Chemical methods include both nondestructive techniques (e.g., the use of detergents and complexing agents, which remove contamination by emulsifying and ion exchange), and destructive techniques (e.g., the use of caustics and acids, which dissolve and corrode contamination and sometimes the base material).

Electrochemical methods are destructive, electrolytically removing contamination and some of the base material. Table 7.2 in Appendix B summarizes the applicability, advantages, and disadvantages of specific methods in each of these broad classes.

B. Aging and Sealing. Aging involves isolating a contaminated object until radioactive decay has reduced the contamination to an acceptable level. This approach is suitable only when short-lived radionuclides are involved. Aging for 10 half-lives reduces the contamination level to one-thousandth (1/1000) of the original level.

Sealing involves fixing radioactivity in place by covering it with an impermeable material such as earth, asphalt, cement, paint, or plastic. Sealing is most effective for alpha and low-level beta-gamma contamination. Most sealants are adequate for shielding alpha and some beta contamination. However, thick, high-density materials (e.g., concrete or several inches of earth) are needed to sufficiently attenuate gamma rays. Sealing is of most value where the primary concern is preventing the spread of relatively low levels of contamination, and where dose rate is not a serious concern. Table 7.3 in Appendix B provides a brief description of methods used for sealing contamination in place.

7.4.2 Selection of Decontamination Methods

The selection and application of decontamination methods is dependent upon the material or equipment to be decontaminated. For extensive decontamination, outside assistance may be necessary. Methods may be used individually or in combination. When more than one method is to be used, the least harsh or abrasive method should be used first. Table 7.4 in Appendix B lists some types of surfaces, materials, and equipment, and identifies methods suitable for decontaminating each. In the case of contaminated commodities, consult the appropriate technical manual for decontamination procedures.

Where extensive decontamination work is to be performed, several methods or combinations of methods can be tested on different areas of the same surface and the results can be compared using the decontamination factor (DF), the commonly used measure of decontamination effectiveness. The DF is calculated as follows:

$$DF = \frac{\text{surface contamination before decontamination}}{\text{residual surface contamination after decontamination}}$$

or

$$DF = \frac{\text{dpm before decontaminaton}}{\text{dpm after decontamination}}$$

The higher the DF, the more effective the method. High DFs are generally achieved with the initial application of any method, but subsequent applications may be less effective. Rinsing usually improves the DF of any decontamination procedure.

All other factors being equal, the decontamination method with the highest DF should be used. However, the resources available for decontamination and the destructiveness of each method also affect the choice of decontamination methods. The RPO should maintain records of the DF obtained during each decontamination in order to assist in the selection of procedures for future decontaminations.

7.4.3 Specific Decontamination Techniques

Specific techniques for decontaminating equipment and materials are described in Appendix C. The techniques include the use of tape patches, vacuum cleaning, wiping or mopping, water jets, detergents, complexing agents, organic solvents, acids, and caustic solutions.

REFERENCES

U.S. Department of the Army, Headquarters. Chemical, Biological and Radiological (CBR) Decontamination. TM 3-2300, Washington, D.C.

U.S. Department of the Army, Headquarters. Safety - Accident Reporting and Records. AR 385-40, Washington, D.C.

APPENDIX A

PERSONNEL DECONTAMINATION PROCEDURES

A.1 Localized Skin Decontamination
A.2 Hair and Scalp Decontamination
A.3 General Body Decontamination
A.4 Facial Decontamination
A.5 Eye, Ear, and Mouth Decontamination
A.6 Nasal Decontamination

APPENDIX A

PERSONNEL DECONTAMINATION PROCEDURES

A.1 LOCALIZED SKIN DECONTAMINATION

Prerequisites

1. Survey to identify extent and magnitude of contamination.
2. Obtain medical assistance if harsh decontamination methods will be necessary.
3. Collect materials needed for decontamination.
4. Document steps and survey results in the appropriate log.

Precautions

1. Medical treatment takes priority over decontamination.
2. Do not spread contamination to clean areas.
3. Do not reuse applicators (replace after each time skin is touched).
4. Handle all waste materials as contaminated waste.
5. Stop decontamination procedures if evidence of skin damage appears or if person complains of soreness or stinging; contact medical personnel for assistance.
6. Person performing decontamination should take precautions not to become contaminated (i.e., wear gloves and other protective clothing as required).

Procedure for Spot Decontamination

1. Press masking tape over contaminated area.
2. Slowly remove and discard.
3. Repeat as necessary, avoiding skin irritation.
4. Proceed with area decontamination if tape method is not effective.

Procedures for Area Decontamination (in increasirg order of harshness)

1. Soap and water

   Use one or more of the following techniques until no further reduction in contamination occurs:

   (a) Wash with mild bar soap and cool or lukewarm water.

   (b) Wash with abrasive soap and water; this method is particularly applicable to toughened skin areas such as fingertips and the palms of the hands.

   (c) Swab with mild liquid soap using cotton-tipped applicators, then rinse with water.

   (d) Use a soft hand brush in combination with any of the above techniques.

   Consult with medical personnel before proceeding with harsher techniques.

2. Detergent and water

   (a) Wash using a detergent and water.

   (b) Make a paste by first lathering the skin area with mild soap and water, then applying detergent powder to lathered skin and working into a paste; rub skin area and rinse paste off.

3. Mild oxidizing agent

   Apply household bleach full strength using cotton sponges or applicators. Rinse after each application. Continue until no further contamination reduction occurs.

4. EDTA solution

   Prepare a 10% EDTA solution by dissolving 10 grams of EDTA salts (Na_4EDTA) in 100 ml of water. (This solution can be prepared in advance and stored.) Apply the solution to the skin using cotton sponges. Rinse after application. Do not apply more than two times.

5. Strong oxidizing agent

   Prepare a saturated solution of potassium permanganate ($KMnO_4$) by dissolving $KMnO_4$ crystals in 1 ounce of water until no more crystals will dissolve (solution will be a dark red or brown). Prepare a saturated solution of sodium bisulfite ($NaHSO_3$) by dissolving $NaHSO_3$ crystals in 1 ounce of water until no more crystals will dissolve. Paint contaminated skin area with $KMnO_4$ solution using cotton applicators or sponges. Allow to dry, then repeat two more times. Remove brown stain by gently swabbing with $NaHSO_3$ solution using cotton swabs. Then rinse with water. If necessary, repeat the application one time.

6. Further decontamination

   If contamination remains after all these procedures have been tried, a medical expert should be consulted for assistance.

7. Post-decontamination

   Following successful decontamination, apply hand lotion to skin to prevent chapping.

8. Sweating

   If soreness or tenderness develops during decontamination, the procedure being used should be stopped for a time. During this interval, the contaminated area can be covered with plastic and allowed to sweat, thus cleansing the area from the inside out. The area should then be gently washed in lukewarm water. (This method is particularly useful for decontaminating the hands, using surgeons' gloves for covering.)

A.2 HAIR AND SCALP DECONTAMINATION

Prerequisites

1. Survey to identify extent and magnitude of contamination.
2. Collect materials needed for decontamination.
3. Document steps and survey results in the appropriate log.

Precautions

1. Medical treatment takes priority over decontamination.
2. Do not spread contamination to clean areas.
3. Do not reuse applicators (replace after each time skin is touched).
4. Handle all waste materials as contaminated waste.
5. Stop decontamination procedures if evidence of skin damage appears or if person complains of soreness or stinging; contact medical personnel for assistance.
6. Person performing decontamination should take precautions not to become contaminated (i.e., wear gloves and other protective clothing as required).

Procedure

1. Contaminated person should remove outer clothing and put on overalls or a laboratory coat and surgeons' gloves.
2. Wrap a towel around the person's neck.
3. Bend the person over a sink or basin and wash hair using mild soap or shampoo. Massage hair and scalp carefully, preventing lather or water from entering the ears, eyes, nose, or mouth.
4. Rinse hair with water. Change the towel if it becomes saturated.
5. Thoroughly dry the hair with towels (do not use a blow dryer).
6. Resurvey hair, also checking face and neck.
7. Repeat shampoo process as long as it is effective.
8. If shampooing ceases to be effective, contaminated hair can be cut with scissor or clippers and the scalp can be decontaminated using the procedures for localized skin decontamination.

A.3 GENERAL BODY DECONTAMINATION

Prerequisites

1. Survey to identify extent and magnitude of contamination.
2. Collect materials needed for decontamination.
3. Document steps and survey results in the appropriate log.

Precautions

1. Medical treatment takes priority over decontamination.
2. Do not spread contamination to clean areas.
3. Do not reuse applicators (replace after each time skin is touched).
4. Handle all waste materials as contaminated waste.
5. Stop decontamination procedures if evidence of skin damage appears or if person complains of soreness or stinging; contact medical personnel for assistance.
6. Person performing decontamination should take precautions not to become contaminated (i.e., wear gloves and other protective clothing as required).

Procedure

1. Remove clothing.
2. Shower with lukewarm water.
3. Lather, using mild soap and soft brush or scrub pad.
4. Rinse, taking care not to spread contamination to skin or body openings.
5. Survey and repeat as necessary.
6. If only localized contamination remains, follow procedures for localized skin decontamination.

A.4 FACIAL DECONTAMINATION

Prerequisites

1. Survey to identify extent and magnitude of contamination.
2. Collect materials needed for decontamination.
3. Document steps and survey results in the appropriate log.

Precautions

1. Medical treatment takes priority over decontamination.
2. Do not spread contamination to clean areas.
3. Do not reuse applicators (replace after each time skin is touched).
4. Handle all waste materials as contaminated waste.
5. Stop decontamination procedures if evidence of skin damage appears or if person complains of soreness or stinging; contact medical personnel for assistance.
6. Person performing decontamination should take precautions not to become contaminated (i.e., wear gloves and other protective clothing as required).

Procedure

1. Use only mild soap and water to decontaminate the face.
2. Exercise special caution to prevent the spread of contamination to eyes, ears, nose, or mouth.
3. Avoid the use of oxidizing agents because of the sensitivity of facial skin and to prevent harm to the eyes.
4. Take nasal smears to assess the presence of nasal contamination.
5. Contact medical personnel for assistance in treating persons with high levels of facial contamination or a suspected internal deposition of radioactivity.

A.5 EYE, EAR, AND MOUTH DECONTAMINATION

Prerequisites

1. Obtain assistance of medical personnel.
2. Survey to identify extent and magnitude of contamination.
3. Collect materials needed for decontamination.
4. Document steps and survey results in the appropriate log.

Precautions

1. Medical treatment takes priority over decontamination.
2. Do not spread contamination to clean areas.
3. Do not reuse applicators (replace after each time skin is touched).
4. Handle all waste materials as contaminated waste.
5. Stop decontamination procedures if evidence of skin damage appears or if person complains of soreness or stinging; contact medical personnel for assistance.
6. Person performing decontamination should take precautions not to become contaminated (i.e., wear gloves and other protective clothing as required).

Procedure for Eye or Ear Decontamination

1. Flush with water. A fountain can be prepared by attaching a flexible tube to a faucet or water bottle.
2. Survey.
3. Repeat as necessary.
4. If eye becomes irritated or activity cannot be removed, obtain further medical assistance.
5. Fluids or agents other than water should not be used unless approved by medical personnel.

Procedure for Mouth Decontamination

1. Special Cautions:

   (a) Under no circumstances should a person with mouth contamination be allowed to eat, drink, chew, or use tobacco until decontaminated.

   (b) In no cases shall oxidizing agents (bleach, potassium permanganate, or sodium bisulfite) be used in the mouth because they will damage the mucous membranes.

2. For localized mouth contamination (spot on tongue or tooth), swab with an applicator or cotton sponge.

3. For general mouth contamination, flush using tap water and a flexible tube connected to a faucet or water bottle (the fountain method).

4. If contamination cannot be effectively removed by flushing, further medical assistance should be obtained.

5. Bioassay should be initiated for individuals with mouth contamination.

A.6 NASAL DECONTAMINATION

Prerequisites

1. Obtain the assistance of medical personnel.

2. Survey to identify extent and magnitude of contamination.

3. Collect materials needed for decontamination.

4. Document steps and survey results in the appropriate log.

Precautions

1. Medical treatment takes priority over decontamination.

2. Do not spread contamination to clean areas.

3. Do not reuse applicators (replace after each time skin is touched).

4. Handle all waste materials as contaminated waste.

5. Stop decontamination procedures if evidence of skin damage appears or if person complains of soreness or stinging; cortact medical personnel for assistance.

6. Person performing decontamination should take precautions not to become contaminated (i.e., wear gloves and other protective clothing as required).

Procedure

1. When nasal contamination is suspected, have the person blow nose into disposable tissue. Survey used tissue and nose.
2. Take smears externally on the nose and upper lip area using filter papers moistened with water.
3. Take smears inside each nostril using cotton-tipped applicators moistened with water.
4. Gently swab nasal passages using wet cotton applicators and periodically have the person blow nose into tissue.
5. If contamination is not removed, obtain further medical assistance in performing nasal irrigation.
6. Bioassay should be initiated for individuals with nasal contamination.

APPENDIX B

EQUIPMENT AND MATERIAL DECONTAMINATION METHODS

Table 7.2. Contamination Removal Methods

Table 7.3. Sealing Methods

Table 7.4. Decontamination Methods for Various Surfaces

TABLE 7.2. Contamination Removal Methods

Method	Applicability	Advantages	Disadvantages
MANUAL CLEANING			
Tape patches	. Dry, localized contamination	. Inexpensive, simple	. Useful only on small areas; can be very time-consuming
Strippable coating	. Dry contamination . Spray on coating, peel off when set	. Similar to tape patches . Better suited for larger surface areas	
Vacuum cleaner (dry)	. Dry surfaces with loose contamination	. Good preparatory step for further decontamination . Effective for dry, porous surfaces	. Vacuum cleaner exhaust must have high-efficiency filter . Contamination concentrated in collection bag can cause dose rate concerns . Not effective for crusted deposits . Airborne radioactivity may be generated
Vacuum cleaner (wet)	. Contaminated liquids . Wet spray or washdown of dry contamination	. Less risk of airborne radioactivity	. Liquid waste generated
Wet wipe or mop	. Dust or accumulated contamination; wipe with water or decon solution . Good follow-up for vacuuming or other methods	. Versatile, simple	. Worker intensive . May involve higher worker dose because of proximity to work
Brushing	. Loose, crusty contamination . Debris	. Preparatory step	. Can generate airborne radioactivity . Not effective for dust or fine particulates

TABLE 7.2. (continued)

Method	Applicability	Advantages	Disadvantages
MECHANICAL CLEANING			
Water jet Steam jet	. Nonporous surfaces (metal, paint, plastic, etc.) . Not suitable for porous surfaces (wood, concrete, etc.)	. May be used in low-pressure hose or high-pressure jet (10,000 psi) applicators . High-pressure jets can be very effective in loosening and dissolving deposits . Can quickly decontaminate large areas	. Can drive alpha contamination into concrete . Drainage must be controlled . Liquid becomes contaminated . Not effective on oiled surfaces . Danger in handling high-pressure nozzles . Airborne contamination probable
Soaking Spraying	. Small and moderate-sized equipment	. Soaking provides good access to surfaces . Many soaking agents available . Spraying combines mechanical action with chemical action	. May require support equipment and systems . Can produce large waste volume
Ultrasonic cleaning	. Small parts	. Combines soaking and mechanical action . Remote operation . Rapid decontamination of irregular shapes and crevices	. Not always useful for strongly adsorbed or absorbed contamination
Vibratory finishing	. Small tools	. Removes rust and gross contamination	. Size limitation . More suitable for exposure reduction than complete decontamination
Freon cleaning	. Cloth, plastic, small tools, respirators	. Remote or manual operation . Small quantity of waste generated	

TABLE 7.2. (continued)

Method	Applicability	Advantages	Disadvantages
ABRASIVE METHODS			
Abrasive blasting	. Irregularly shaped or large surfaces where critical dimensions are not involved	. Rapid, very effective decontamination . Variety of abrasives available . Wet or vacuum blasting can reduce generation of airborne activity and spread of contamination	. Usually generates airborne contamination . May spread surface contamination . Grit size must be finer than surface finish
Grinding	. Small objects or isolated spots on large objects	. Economical, effective	. Leaves residual contamination . Airborne contamination
Chipping Spalling Cutting	. Concrete or structural surfaces	. Effective for removal of porous surfaces . Removes surface in thin layers (0.3 cm per pass)	. Leaves residual contamination . Can generate airborne activity . Usually slow
Scabbling	. Concrete surfaces	. Same as for chipping and spalling . Faster than chipping and spalling . Can be fitted with high-efficiency filter	. Can leave residual contaminants . Can generate airborne activity

TABLE 7.2. (continued)

Method	Applicability	Advantages	Disadvantages
CHEMICAL CLEANING[a]			
Detergents	Nonporous surfaces with contaminated films	. Dissolves contaminated films and oils . May be applied by rag or soaking	. Relatively mild; may not be effective for deep-seated contamination
Complexing agents (oxalates, carbonates, citrates, EDTA)	Nonporous surfaces (unweathered, no rust)	. Contamination retained in solution . Easily stored . Carbonates and citrates are nontoxic, non-corrosive	. Little penetrating power . Not effective for weathered surfaces
Organic solvents	Nonporous surfaces (greased, waxed, painted, plastic-covered)	. Quick dissolving action . Solvent can be recovered by distillation	. Flammable . Ventilation required . Toxic . Not as effective as acid processes
Inorganic acids	Metal surfaces (porous deposits, rust, corrosion)	. Corrosive action on porous deposits	. Personnel hazard . Toxic . Explosive gases generated . Ventilation required

(a) Many chemical solutions can be applied either hot or cold. Hot applications are usually more effective than cold applications.

TABLE 7.2. (continued)

Method	Applicability	Advantages	Disadvantages
CHEMICAL CLEANING (continued)			
Acid mixtures	Nonporous surfaces (porous deposits)	. Highly effective dissolving action	. Weathered surfaces may require long treatment
Caustics	Painted surfaces (horizontal)	. Softens paint with minimum contact . Easy storage	. Personnel hazard . Slow reaction rate . Not efficient for vertical or overhead surfaces . Do not use on aluminum or magnesium
Trisodium phosphate	Painted surfaces (vertical and overhead)	. Softens paint	. Destructive to paint . Do not use on aluminum or magnesium
ELECTROCHEMICAL CLEANING			
Electropolishing	Small tools and parts, tanks, pipes, larger surfaces	. Highly effective . Can be aimed at spots . Fast decontamination . Remote application	. Removes thin layer (2 mils) of base material . Attacks high spots first . Possible material compatibility problems with acid electrolytes

TABLE 7.3. Sealing Methods

Method	Applicability	Advantages	Disadvantages
Earth	. Temporary barrier	. Material readily available . Shovel application small area	. Heavy equipment needed for large areas
Asphalt Concrete	. Roads, surfaces	. Thin layer provides quick temporary seal . Thicker layer (2.5 cm) provides permanent seal . Complete alpha and beta shielding	. Heavy equipment needed
Grout	. Concrete or masonary surfaces	. Can be spread on by hand . Thin layer (0.6 cm) provides effective seal	
Paint Varnish Plastic	. Areas subject to periodic recontamination	. Inexpensive . Can be sprayed or brushed on . Effectively shields alpha and low-energy radiation	. Degrades with time . No gamma attentuation

TABLE 7.4. Decontamination Methods for Various Surfaces

Surface	Description	Decontamination Methods
Terrain	Decontamination of terrain usually involves large areas and requires elaborate machinery or extensive manpower. Fields and vegetation strongly absorb liquids, making wet decontamination procedures impractical. Natural objects, loose dirt, and dust make removal procedures such as brushing or vacuuming difficult.	Earth moving (removal) Sealing in place
Hard, porous surfaces (concrete, asphalt, brick, stone, wood)	Porous surfaces absorb or trap liquids within the pores in the material. Any radionuclides dissolved or suspended in the trapped liquid remain in the pores after the liquid evaporates. Porous materials also mechanically trap dust particles. Complete removal of any contamination is difficult.	Vacuum cleaning Water flush Destructive removal (planing, chipping, scabbling)
Fibrous material (cloth, canvas)	Fibrous materials contain many small pores, allowing free flow of liquids through them. There is little chance for particles to become permanently lodged, absorbed, or adsorbed in the pores.	Brush Tape Patches Laundering Freon
Glass Porcelain Plastic	Glass, metal, plastic, and porcelain surfaces, which are generally smooth and relatively inert to chemicals, are best decontaminated by using water or water and detergent. Dust and liquids have little chance of becoming trapped on these surfaces. Adsorbed materials are readily removed with slight abrasion or brushing. Glass is attacked and etched by strong caustics. Plastics may be attacked by caustics, oxidizing and mineral acids, and organic chemicals.	Water Detergent Complexing agents Inorganic acids Freon
Metal	Surface description similar to glass, porcelain, and plastic. Metal is attacked and dissolved by strong oxidizing agents and acids.	Water Detergent Complexing agents Organic acids Inorganic acids Acid mixtures Oxidizing agents Ultrasonics Electropolishing Vibratory finishing Freon

TABLE 7.4. (continued)

Surface	Description	Decontamination Methods
Paint Varnish	Surfaces that are painted, varnished, or waxed are generally smooth and nonporous. Dust and liquids are readily removed by wiping, brushing, or vacuuming. Adsorbed materials are usually removable by water, detergent, or complexing agents. These surfaces do not stand up to heavy abrasive techniques. If they become contaminated, they may be removed by caustics, acids, and organic chemicals.	Vacuum cleaning Wiping Water Detergent Complexing agents Organic solvents Caustics Abrasion
Floor coverings: Linoleum Tile	Floor coverings such as linoleum and asphalt tile are best decontaminated by wiping, brushing, or vacuuming. The surfaces readily absorb liquids unless protected by a layer of wax, which limits absorption. Because they are smooth, they do not mechanically trap solid particles; however, cracks between tiles can absorb some contamination. Tile and linoleum are attacked by strong acids, caustics, and organic solvents.	Wipe Brush Vacuum Destructive removal Seal
Rubber	Rubber is a porous material that strongly absorbs liquids. It is not easily decontaminated by abrasive techniques. Brushing and vacuuming remove surface contamination, and water and detergents remove some absorbed contamination. Strong acids, alkalies, and organic solvents deteriorate and decompose rubber.	Brush Vacuum Wipe Detergent
Leather (shoes)	Leather is a porous material that can be very difficult to decontaminate. Scraping and tape patches are most effective.	Tape patches Knife and sandpaper Acetone wipe

APPENDIX C

EQUIPMENT AND MATERIAL DECONTAMINATION PROCEDURES

C.1 Tape Patches
C.2 Vacuum Cleaning
C.3 Wiping or Mopping
C.4 Water Jets
C.5 Detergents
C.6 Complexing Agents
C.7 Organic Solvents
C.8 Acids and Acid Mixtures
C.9 Caustics

APPENDIX C

EQUIPMENT AND MATERIAL DECONTAMINATION PROCEDURES

C.1 TAPE PATCHES

Materials

1. Masking, adhesive, friction, or duct tape

Procedure

1. Place tape over contaminated area.
2. Remove tape and discard as radioactive waste.
3. Repeat as long as effective.

C.2 VACUUM CLEANING

Materials

1. Conventional wet or dry vacuum cleaners may be used if modified to include a high-efficiency particulate air (HEPA) filter on the exhaust.

Procedure

1. Use conventional vacuum-cleaning techniques.
2. Periodically monitor build-up of radioactivity or dose rate from bag or canister during operation.
3. Dispose of bag or collection container as radioactive waste.
4. For extensive use, monitor build-up of dose rate from collection container and HEPA filter.

C.3 WIPING OR MOPPING

Materials

1. Mop, cloth, or towel.

Procedure

1. Wipe or wet-mop using a decontaminating agent and hot water.
2. Rinse with clean water, damp-mopping.
3. Repeat as necessary.

C.4 WATER JETS

Materials

1. High-pressure, low-volume jet and/or low-pressure jet or spray.

Procedure

1. Spray from top to bottom at an angle of 30° to 45°.
2. Use high-pressure jets to loosen decontamination.
3. Use low-pressure jets or sprays to wash and flush.
4. Determine cleaning rate experimentally or else use 0.5 to 0.9 m^2/min.

C.5 DETERGENTS

Materials

1. Detergent.

Procedure

1. Apply full strength or per manufacturer's recommendations.
2. Wipe with towel or rag.
3. Powered brush may be used.
4. May be applied by a mist applicator, using caution to prevent spread to other surfaces.

C.6 COMPLEXING AGENTS

Materials

1. Solution containing 3% (by weight) of complexing agent (e.g., EDTA).

Procedure

1. Spray surface with agent.
2. Keep moist for 30 minutes.
3. Flush with water.

   Note: May be applied to vertical and overhead surfaces by adding chemical foam (sodium carbonate or aluminum sulfate).

C.7 ORGANIC SOLVENTS

Materials

1. Kerosene, paint thinner, or acetone.

Procedure

1. Use standard wiping techniques.
2. Immerse in solvent bath.

   Caution: High flammability and toxic fumes. The use of acids and complexing agents is generally preferable.

C.8 ACIDS AND ACID MIXTURES

Materials

1. Single Acids (1 to 2 normality)
   3%-6% sulfuric acid
   9%-18% hydrochloric acid
   5% oxalic acid
2. Acid Mixture
   0.4 liter hydrochloric acid
   90 grams sodium acetate
   4 liters water
3. Other acid mixtures may include acetic acids, citric acids, acetates, citrates.

Procedure

1. Use dip bath for movable items.
2. Leave weathered surfaces in contact with acid solution for 1 hour.
3. Allow pipe circulation systems to soak for 2 to 4 hours.
4. Flush with water.
5. Flush with neutralizing solution.
6. Flush with water.

   Caution: Personnel hazard, toxic and explosive fumes generated. Provide good ventilation.

C.9 CAUSTICS

Materials

1. Lye (sodium hydroxide)
2. Calcium hydroxide
3. Potassium hydroxide
4. Typical solution for removing paint:
   38 liters water
   1.8 kg lye
   2.7 kg boiler compound
   0.34 kg cornstarch

Procedure

1. Apply caustic solution to painted surface.
2. Keep solution in contact with paint until paint is soft enough to be washed off with water.
3. Wash off paint and caustic solution with water.
4. Remove remaining paint with scraper.

   Caution: Caustics pose personnel burn hazard.

CHAPTER 8. SELECTION AND DESIGN OF RADIATION FACILITIES

FIGURES

CHAPTER 8. SELECTION AND DESIGN OF RADIATION FACILITIES

Facilities in which radioactive materials are used have specific needs that must be recognized and planned for from the initial design phase through the construction and operation of each facility. The location of the facility must be considered in relation to the work that will be carried on there. The building must be designed to keep radioactive materials in certain areas while still allowing efficient operation. Finally, equipment must be built in or brought in to control external and internal radiation doses to personnel and to keep the amount of radioactive material leaving the facility within permissible limits.

The purposes of this chapter are: 1) to help the Radiation Protection Officer (RPO) and the Ionizing Radiation Control Committee (IRCC) judge whether a facility is adequate for handling radioactive materials, and 2) to delineate what should be considered when a facility is being designed and the rationale behind each item. Because DARCOM and the installation's engineering staff have ultimate responsibility for facility design, this chapter is for information purposes only.

Section 8.1 GENERAL PRINCIPLES

Safety should be achieved as much as possible through engineered safeguards rather than administrative controls or the use of personnel protective equipment. The National Council on Radiation Protection and Measurements (NCRP) recommends in Report No. 59 (1978) that a qualified expert be consulted during the planning and design of new and modified radiation facilities to ensure the incorporation of proper radiation safety procedures. Certification by the American Board of Health Physics or the American Board of Radiology is evidence of a consultant's qualifications.

Items that must be considered when a new facility is being planned or an existing structure is being renovated include:

1. meteorological and hydrological parameters of the site
2. facility layout, which should be compatible with the establishment of contamination areas
3. shielding, especially with respect to floor-loading limits
4. ventilation, which should be capable of controlling the movement of air to prevent or minimize the spread of contamination within the facility
5. types of monitoring equipment needed.

The facility should be arranged to meet the following objectives:

1. keep dose equivalents received by personnel as low as is reasonably achievable (ALARA)
2. confine radioactive materials accidently released within the facility and control releases from the facility to levels below the concentration guides in 10 CFR 20, Appendix B, Table II, averaged over 2 hours
3. achieve a uniform level of safety through physical and engineered safeguards
4. accommodate normal or anticipated changes in mission requirements without compromising radiation protection.

Section 8.2 INITIAL PLANNING PROCESS

The terms "facility design," "radiological design," and "radiological engineering" are often used interchangeably, although they have different meanings. Design is the planning and development of a facility as opposed to its actually construction and operation. Facility design refers to a plan for a building or installation as a whole, and thus includes nonradiological as well as radiological design features. Radiological design refers to the specific set of design features included because of the planned presence of radioactivity or radiation-generating machines. Radiological requirements should be made known to the architect and/or engineer responsible for designing a facility as early as possible, to minimize the cost of incorporating safety features; it is

less expensive to redraw preliminary plans than final blueprints, and less expensive to revise final blueprints than to rebuild or alter the finished facility.

Radiological engineering refers to the implementation of the radiological design (i.e., the actual construction). Radiological engineering requires the use of quality control procedures during construction. For example, precautions should be taken to minimize air pockets in concrete walls used for shielding, to sufficiently overlap lead sheets used for shielding, and to ensure that foundations, footings, and pilings have sufficient loadbearing capacity so that concrete shield walls do not buckle or crack. In essence, good radiological engineering ensures that the design criteria are met (Kathren and Selby 1980).

Review of the radiological hygiene aspects of blueprints, drawings, and other documents relating to the design of facilities and devices for generating radiation should be coordinated through channels with the DARCOM Field Safety Activity and the U.S. Army Environmental Hygiene Agency (USAEHA). Therefore, contact with DARCOM and the agency should be made early in the planning process to avoid the necessity of expensive changes in the structural design.

8.2.1 Designs for New Facilities

When a facility is being designed, all proposed uses and needs of the facility--both current and projected--must be considered, especially if the projected needs will exceed the current needs. If possible, the facility should be designed to meet the maximum needs, because the cost of altering or rebuilding may be greater than the cost of overbuilding initially. The scope of work to be performed in the building should be defined in terms of the purpose of the work, the proposed inventories of radioactive materials, the presence of radiation-generating devices, and the expected lifetime of the building.

Many safety features must be considered early in the design of a facility. With few exceptions, shielding and facility layout are difficult to change, and adequate safety often cannot be ensured in a redesigned or rebuilt facility

without high costs and the loss of usable work space. Thus, future uses of the facility, which may include increased workloads, must be considered so that shielding, containment, confinement, and work spaces can be designed to suit those uses.

A. Commander's Responsibility. The local commander is responsible for providing proper facilities for the use of radioactive materials (AR 385-11). Therefore, the commander shall provide for the review and approval of all blueprints, drawings, and other documents relating to the design of facilities that will contain radioactive materials. Assistance in judging the adequacy of new and renovated facilities may be obtained from USAEHA and the DARCOM Field Safety Activity.

B. Ionizating Radiation Control Committee. The IRCC should have as part of its responsibility helping to design safe facilities. The committee should include construction or general engineering personnel and representatives from maintenance, operations, health, and safety, including the RPO. The committee should be informed of all proposed uses for each building, both immediate and future. The local commander shall establish an approval process that guarantees that all safety-related concerns (both radiological and nonradiological) have been addressed and adequately resolved.

(1) Maintenance and Operations Representatives. Representatives from maintenance and operations should be consulted because they are usually aware of the problems associated with various building designs. They can advise on whether a design will allow ease of maintenance and repair, which can minimize work times in radiation areas.

(2) Health and Safety Representatives. The RPO and the other health and safety representatives should be responsible for the following:

1. reviewing the general layout of the facility, giving particular attention to corridors, traffic patterns, radiation areas, change rooms, radiation-monitoring sites, and personnel decontamination facilities
2. working with the installation's environmental coordinator to prepare or coordinate the preparation of the environmental impact statement (if any)

3. identifying manuals and standards that deal with radiological aspects of the facility design

4. ensuring that the ventilation system will provide the public and site personnel with maximum protection against airborne contamination

5. ensuring that maximum practical control of liquid, solid, and gaseous wastes is provided, to protect the environment

6. verifying that the proposed design and application of hoods, glove boxes, and shielded cells ensures ease of decontamination and remote operation, to reduce occupational exposures

7. ensuring that the thickness of all shielding meets design criteria, and coordinating shielding calculations and design to keep radiation doses ALARA

8. ensuring that needs for sampling and monitoring instrumentation have been identified and that the instrumentation being provided meets the latest occupational and environmental standards, can be installed properly, and is capable of obtaining representative samples

9. ensuring that radiological safeguards and safety systems are adequately protected from fires, floods, and other similar accidents, and are fail-safe

10. assessing the adequacy of facilities for receiving, storing, and packaging any radioactive wastes that may be produced during the operation of the building.

8.2.2 Review of Designs for Modifying Existing Facilities

How extensively a facility is being modified influences the extent of the design review needed. Major modifications, such as extensive renovation of a radioactively contaminated facility or preparation of a facility that has never before housed radioactive materials, may require application of all steps involved in the design of a new facility and may therefore require the same attention from the members of the IRCC. The RPO, or the appropriate health and safety representative, has the following additional responsibilities whenever an existing radiation facility is being upgraded:

1. If the building previously contained radioactive material, evaluate the modification plans to ensure that radiation dose equivalents received by construction workers during the renovation are kept ALARA. (Consider removing radioactive sources and decontaminating the facility.)
2. Evaluate the impact of the modification on existing safety systems, such as air filters and ventilation systems.
3. Review the design of any structures needed to contain radioactive materials (e.g., greenhouses and special waste containers).
4. Approve all modifications.

Section 8.3 SITE SELECTION

The initial step in selecting the site for a radiation facility is to establish the requirements of the facility and the interrelations between the facility and its environment. Proposed sites and the area surrounding each should be reviewed for location and for distances from air, ground, and water traffic, pipelines, and fixed manufacturing, processing, and storage facilities.

8.3.1 Impact of Surrounding Operations Upon the Proposed Facility

The level of background radiation at a proposed site can affect some operations and should be considered during site selection. Other external factors affecting site selection are the location of other facilities, the potential for fires, explosions, and chemical spills, and any need for restricted access.

A. Background Radiation. Background radiation is an important consideration for facilities that will house laboratory counting instruments, which are extremely sensitive to radiation. Fluctuations in the level of background radiation can affect the instrument readings, and high background radiation rates, even if they are constant, increase the lower limit of detection for these instruments.

Background radiation levels can be increased by either natural or man-made sources. One natural source of increased background radiation is certain types of rocks that have a fixed radiation. The evolution of radon gas from rocks and soil also raises the concentration of radioactive material in air and thus results in more surface contamination and higher dose rates. Man-made causes of increased background radiation levels include nuclear power reactors and mining and milling operations. Uranium mining is an obvious cause; however, phosphate mining and even coal mining are also sources of background radiation.

The extent to which radiation background levels fluctuate because of these sources is small and under ordinary circumstances does not present a significant radiation hazard to personnel. However, a radiation hazard may occur in submerged or underground facilities, especially if the air flow rates are low.

B. Effluents From Facility and Nearby Operations. A facility should be designed so that its air intakes are not likely to draw in its own exhaust materials. As a general rule, air intakes should be at the upwind end of the facility and exhaust vents should be at the downwind end, with the prevailing wind direction used as a guide. In addition, air intakes should be at least 155 meters away from the exhaust vent of any other facility that is venting radioactive material or other toxic or hazardous materials.

C. Fire and Explosion Hazards. Operations that might present fire and explosion hazards include petroleum refineries and storage facilities, docking facilities (for example, for oil tankers), and chemical-manufacturing plants. Also, military depots may be sites of storage for explosive compounds. Radiation facilities should be located at a safe distance from such hazards.

D. Chemical Spills. The manufacture, storage, and transportation of chemicals lead to the potential for chemical spills or releases. The release of toxic gas may require that a facility be evacuated promptly. However, in some facilities such as nuclear reactors, operators cannot be evacuated immediately. In such cases, protection must be provided for the workers.

E. Access Control. Access to a facility may be restricted for either radiation safety or national security reasons. Access control for national

security purposes is beyond the scope of this manual. Whenever a high-radiation area is not mechanically secured to prevent unauthorized entry, a guard must be posted (DARCOM-R 385-25). Physical safeguards that are appropriate for the hazard or security classification must be used.

8.3.2 Impact of Proposed Facility Upon Surrounding Area

The use of radioactive materials at a facility may increase the level of background radiation if any materials used outside of sealed containers are released to the environment. The releases may be of two types: routine low-level releases and accidental releases that could be of any magnitude. The possibility of such releases influences the selection of a facility site. The anticipated use of the land around a proposed radiation facility should also be considered in site selection.

A. Potential Environmental Releases. Routine releases usually enter the air from hood vents and enter sanitary sewage systems via floor and sink drains. Radioactive material may also be transported to the environment on the clothing of personnel and can be tracked about extensively if it gets on their shoes. Facilities in which radioactive materials are used should be located downwind from major metropolitan areas and in flat or gently rolling terrain, so that any radioactive material accidentally released into the air is dispersed rapidly and evenly, with minimal impact. In addition, engineered safeguards should be provided to prevent or at least limit the release of radioactive materials to the environment. Such safeguards are discussed later in the chapter.

B. Accident Analysis. The potential for accidents should be analyzed before any accident occurs. The RPO and individuals familiar with ventilating systems should review the proposed levels of radioactivity in each laboratory. Accidents that could result in the release of radioactive materials should then be analyzed. This analysis can be detailed, involving determination of the possible causes, probabilities, and impacts of an accident; or it can be as simple as assuming that the largest amount of material that might be unsealed at any time is available for release (see Chapter 11).

Accident analysis is extremely important for major facilities. It can help ensure that engineered safeguards are provided to prevent or minimize radiation exposures of personnel and the public. If an accident analysis is performed early in the design process, safeguards may be suggested that otherwise would have been omitted.

C. Future Land Use. Sources of information on projected population growth and proposed land uses should be consulted. County engineers can provide information on public roads and traffic volumes; local government councils may have information on population growth, proposed new industries, or future transportation routes; and zoning boards are sources of information on land use controls. The increase in the local population brought about by the construction and use of the proposed facility should also be considered, as it may not have been included in the projections of the state and local agencies just mentioned.

D. Additional Considerations. Before a particular site is selected, the following topics should be considered:

1. personnel traffic routes and their relation to the flow patterns for exhaust air where accidental or routine releases of radioactive material could occur (radioactive material should not be vented to high-traffic areas)
2. the relationship between the exhaust and air supply systems of various facilities (radioactive material should not be vented where it is likely to be drawn into other buildings or back into the building it came from)
3. the impact of additional radioactive waste on waste removal systems (e.g., consider stress on sewer systems that may contain radioactive material, on retention or diversion systems, and on systems that handle liquid waste containing high levels of radioactivity)
4. the availability of emergency systems (fire, ambulance, and radiological-emergency response teams)
5. the ability to simultaneously evacuate all neighboring facilities

6. the need for special transportation capabilities (railroad spurs, rigs for moving heavy material)

7. the impact of future modifications.

8.3.3 Natural Phenomena

Facilities should be designed to withstand the influences of natural phenomena. This requirement can be relaxed for facilities located where the only natural phenomena likely to occur are those that can be accurately forecasted, such as hurricanes and floods. In these cases, adequate warning time for securing materials and evacuating personnel can be provided. Other phenomena, such as earthquakes, volcanic eruptions, and floods caused by dam failure, cannot be adequately forecasted and may occur with little or no warning.

A. Regional Climate. Meterological conditions that may affect a facility include hurricanes, tornadoes, water spouts, thunderstorms, lightning, hail, and high levels of air pollution.

Data on severe weather phenomena should be based on standard meteorological records from a nearby National Weather Service station or from military or other stations that are recognized as standard installations and that have kept records for a long time.

B. Hydrology. The hydrology of a site should be reviewed, especially if a facility will house large quantities of special nuclear materials (plutonium or uranium enriched in isotope 233 or 235, or any material artificially enriched by either isotope). The hydrologic characteristics of streams, lakes, shore regions, and existing or proposed water control structures (e.g., dams and irrigation ditches) should be considered as they relate to potential flooding of the structure. The hydrology of both surface water and ground water should also be considered as it relates to the possible contamination of these waters by activities within the facility.

C. Geologic and Seismic Considerations. Ideally, the site should be in a geologically stable area--one low in seismicity, free of active faults, underlain by competent foundation materials, and free from the adverse effects of other geologic hazards.

Section 8.4 FACILITY DESIGN

A properly designed facility can lead to reduced radiation doses to personnel through the establishment of designated areas for the use of radioactive materials, and of designated types of laboratories within these areas. The materials used in the construction of a facility and the ease of access to areas within the facility also affect radiation safety for personnel.

8.4.1 General Considerations in Facility Design

The layout of rooms, corridors, entrances, exits, ventilation systems, and other utilities in a building should be designed to meet the following objectives:

1. Keep the dose equivalent received by personnel ALARA.
2. Confine radioactive materials accidentally released within the facility and control any releases from the facility so that they remain below the concentration guides in 10 CFR 20, sections 20.106 and 20.303.
3. Accommodate routine programs or anticipated program changes without compromising radiation protection.

The flow of people and materials in a facility is a function of building design. One design, shown in Figure 8.1, has a central service corridor for equipment, piping, and waste handling. Laboratories on both sides open to both the central corridor and the outer corridors, with offices located between the outer corridors and the outside of the building. The advantages of this design are that it allows for two exits from each laboratory, permits easy access to utilities for the laboratories, and allows radioactive materials to be transferred without affecting the clean areas of the facility. An alternate design might have offices located in one part of the building and laboratories in another, so that only laboratory personnel need enter the laboratory areas.

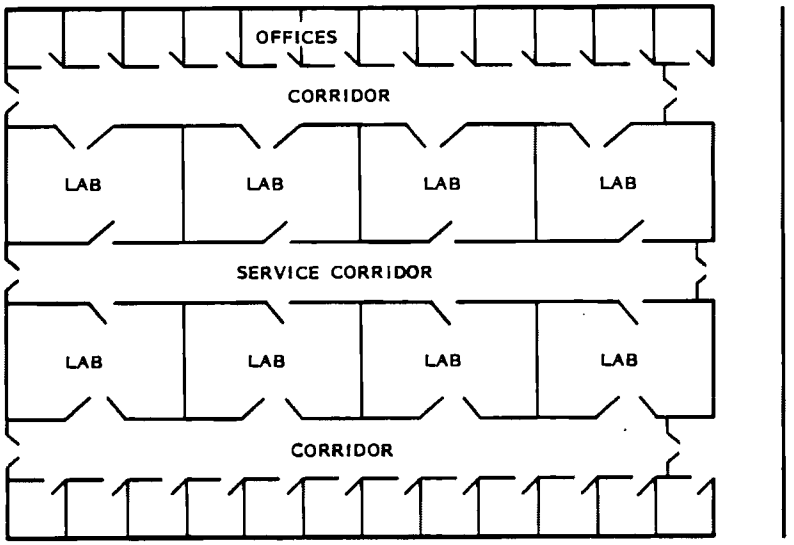


FIGURE 8.1. Facility Layout

8.4.2 Building Areas

Facilities are generally divided into a series of sequential areas that are based upon the presence of radiation or radioactive materials and are designed to control personnel exposure to radiation. The three types of areas--radiation, controlled, and uncontrolled--are described below.

A. Radiation Area. Radiation areas include three subclassifications: radiation areas, high-radiation areas, and airborne-radioactivity areas. A radiation area is defined in 10 CFR 20 as any area accessible to personnel in which radiation levels could result in a major portion of the body receiving a dose-equivalent rate in excess of 5 mrem in any 1 hour or 100 mrem in any 5 consecutive days. For practical purposes, AR 40-14 defines this as any area in which the dose-equivalent rate is greater than 2 mrem/hr but less than 100 mrem/hr. A high-radiation area is any area accessible to personnel in which radiation levels could result in a major portion of the body receiving a dose equivalent in excess of 100 mrem in any 1 hour. All radiation areas must be marked and posted as described in 10 CFR 20.20. An airborne-radioactivity area is any room, enclosure, or operating area where the concentration of airborne radioactivity exceeds the amounts specified in 10 CFR 20, Appendix B, Table I, Column 1 or where the concentration, when averaged over the number of

hours in any week an individual works in the area, will exceed 25% of the amounts specified in 10 CFR 20, Appendix B, Table I, Column 1.

To ensure that regulatory and administrative limits are not exceeded, consideration should be given during facility design to the establishment of radiation areas at:

1. any location where unsealed (unencapsulated) radioactive materials will be stored, handled, or processed
2. any area containing a radiation-generating device
3. any routinely occupied area where an individual would be expected to receive more than 500 mrem in 1 year
4. any area, regardless of the expected occupancy, there the anticipated dose-equivalent rate exceeds 2 mrem/hr
5. any routinely occupied area where the concentration of airborne radioactive materials may exceed 25% of the values presented in 10 CFR 20, Appendix B, Table I, Column 1
6. any area, regardless of the occupancy, where the concentration of airborne radioactive materials may exceed the values presented in 10 CRF 20, Appendix B, Table I, Column 1.

Radiation areas should be remote from offices, lunchrooms, and conference rooms, to preclude the exposure of support personnel (e.g., secretaries and clerks). Persons entering a radiation area should pass through a controlled area. To keep nonradiation workers out of radiation areas during the normal course of their work, separate corridors should be provided.

B. Controlled Area. A controlled area is any area to which access is controlled and in which occupancy and working conditions are controlled for the purpose of protecting personnel against exposure to radiation. Such areas include:

1. any area normally free of contamination that is adjacent to a radiation area and that may become contaminated through accidental spreads or releases from the radiation area

2. any area that may occasionally contain radioactive material because of the transportation of radionuclides between radiation areas or the maintenance of contaminated process equipment that cannot be entirely placed inside a radiation area

3. any area where the anticipated dose-equivalent rate exceeds 0.2 mrem/hr but is less than 2 mrem/hr

4. any area where the concentration of airborne radioactive materials may exceed 50% of the values presented in 10 CFR 20, Appendix B, Table II, Column 1.

C. Uncontrolled Area. An uncontrolled area is any area where direct radiation exposure is not necessary or anticipated in the performance of a job. These areas include "cold" laboratories (those containing no radioactivity), offices, lunchrooms, conference rooms, and reception areas. The traffic patterns in a building should keep radioactive materials from being brought into uncontrolled areas for any reason (such as by delivery personnel). Further, the building should be designed so that the dose-equivalent rate in uncontrolled areas does not exceed 0.2 mrem/hr.

8.4.3 Work Stations

Work stations are subdivisions of a radiation area. One method of designating work stations is to define three classes of laboratories, A, B, and C, which depend upon the radiotoxicity, dispersibility, and total quantity of unsealed radioactive materials to be used. (See Chapter 1, Section 1.6.2, for definitions of the levels of dispersibility, and Chapter 1, Table 1.10, for groupings of radionuclides by degree of radiotoxicity.)

A. Class A Laboratories. Class A laboratories are specially designed and equipped for the safe handling of 1) large quantities of highly radiotoxic materials (groups VI through VIII in Table 1.10) in any dispersible form and 2) large quantities of moderately radiotoxic materials (Groups III through V) in highly dispersible form.

Each Class A laboratory should be wholly within a radiation area and should be separated from uncontrolled areas by at least two confinement barriers (see Figure 8.2). Within a Class A laboratory, a fume hood should be

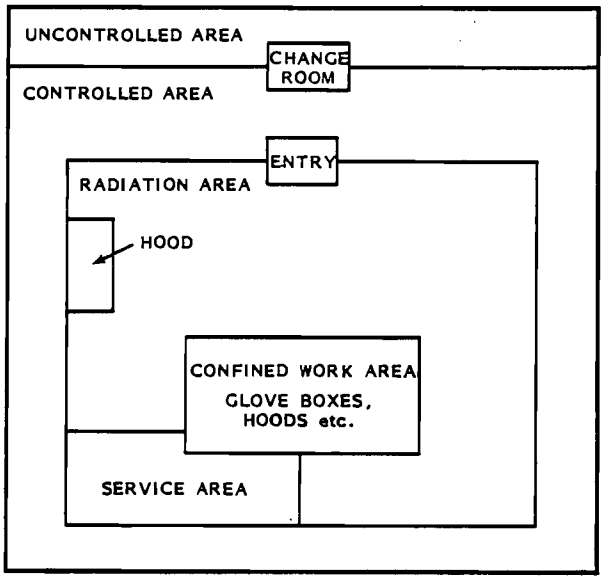


FIGURE 8.2. Class A Laboratory

used for work involving dispersible material, and sealed glove boxes, hot cells, or similar devices should be used for work involving readily or highly dispersible materials. Class A laboratories should have access to a clothing change room through which personnel pass before entering an uncontrolled area.

The air of a Class A laboratory should be exhausted through two stages of high-efficiency particulate air (HEPA) filters that are testable using a dioctylphthalate (DOP) mist. (Because DOP is a suspected carcinogen, it should be used with care.) The air of hoods, glove boxes, or other sealed enclosures where readily and highly dispersible materials are used should be exhausted through three stages of HEPA filters, at least two of which must be DOP-testable. The use of gaseous materials (wet operations) may cause early failure of HEPA filters. Therefore, if these materials are used, HEPA filters may need to be replaced frequently, air flow monitors should be used, and additional filtration devices may be needed.

B. Class B Laboratories. Class B laboratories are designed for the handling of 1) large quantities of minimally radiotoxic materials (Groups I and II in Table 1.10) or 2) moderate quantities of moderately or highly

radiotoxic materials (Groups III through VIII). The materials may range from dispersible to highly dispersible.

Each Class B laboratory should be separated from uncontrolled areas by at least two confinement barriers (see Figure 8.3). A glove box or other enclosure should be used for work with highly radiotoxic or highly dispersible materials. Each laboratory should have at least one fume hood.

The air of a Class B laboratory should be exhausted through at least two DOP-testable HEPA filters that are in series. The exhaust system for hoods, glove boxes, or other enclosures should contain two stages of DOP-testable HEPA filters.

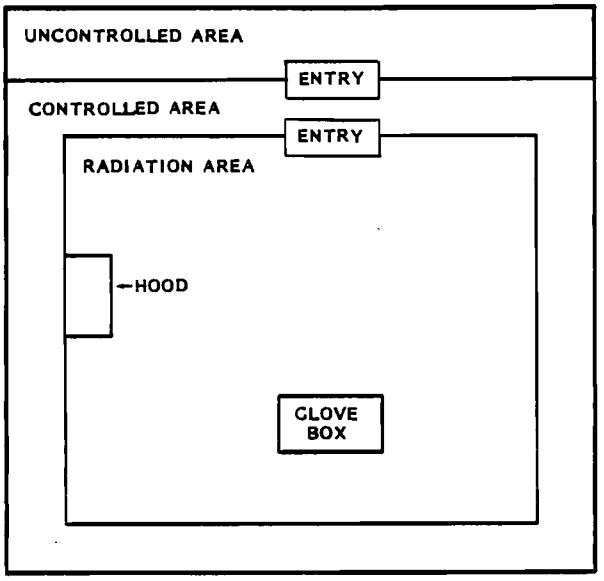


FIGURE 8.3. Class B Laboratory

C. Class C Laboratories. Class C laboratories are designed for work involving simple chemical processes and minimal quantities of radioactive material. Materials of low and moderate radiotoxicity (Groups I through V in Table 1.10) may be present in forms that are dispersible or of limited dispersibility.

Each Class C laboratory should be separated from uncontrolled areas by at least one confinement barrier, which may be the laboratory wall (see Figure 8.4). At least one hood should be provided in each laboratory. The exhaust system should contain at least a single-stage DOP-testable HEPA filter.

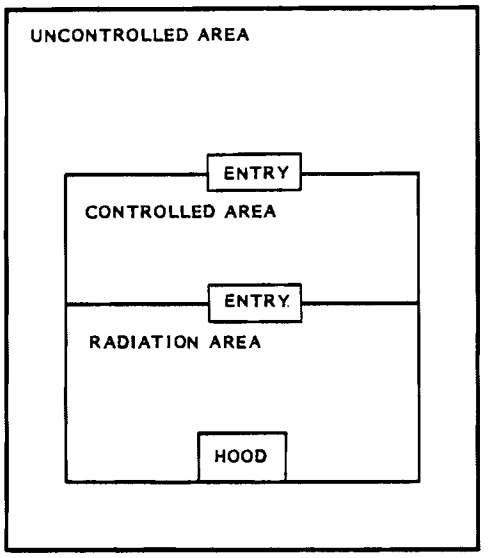


FIGURE 8.4. Class C Laboratory

8.4.4 Building Materials

The building materials used in a radiation facility should be easy to decontaminate, extremely durable, corrosion resistant, and fire resistant. Unfortunately, very few materials combine all of these characteristics.

A. Ease of Decontamination. The building materials chosen should be nonporous and should have few, if any, cracks. They should be readily removable if contaminated, and chemically inert to reduce the likelihood that contamination would become chemically bonded to the materials. (See Chapter 7 for details on the ease of decontaminating various materials.)

(1) Flooring. Flooring materials should be chosen based on price; availability; ease of installation, service, and maintenance; chemical inertness; and any special requirements imposed by the use of radioactive materials. Porous materials such as concrete and wood are not acceptable by themselves; they must be covered by other, removable materials to facilitate decontamination in the event of an accident. Examples of acceptable covering materials include sheet flooring (such as vinyl flooring) or poured vinyl or epoxy floor covering. The floor covering should be sealed and waxed regularly.

(2) Walls and Partitions. Walls and partitions should be protected by coatings that are hard, smooth, and easy to clean. If extensive contamination is possible, then strippable coatings should be used. These form an effective seal over porous wall, ceiling, or floor materials and are easily stripped off or removed when the surface must be decontaminated.

(3) Bench Tops and Laboratory Equipment. Laboratory benches with synthetic or plastic tops are now available. Many of these tops are quite impermeable and durable; consult manufacturers' literature for details. Laboratory equipment can be tested for susceptibility to contamination and ease of decontamination, as described by Fitzgerald (1969). In general, furniture in laboratories where low and intermediate levels of radiation are used should be of high-quality, impermeable materials.

B. Corrosion Resistance. Bench tops, hoods, walls, and floors should be corrosion resistant because the pitted surfaces caused by corrosion are difficult to decontaminate.

C. Fire Resistance. Laboratory facilities should be fire resistant. Where a fire could result in the dispersal of radioactive materials, exits and a means of closing the facility to prevent the spread of radioactive materials should be provided. Fire extinguishers should be located throughout each facility, and showers and fire extinguishers should be provided in laboratories where flammable chemicals are used.

8.4.5 Building Access

Consideration should be given to pathways for moving radioactive materials in and out of buildings and laboratories. Examples of items that should be considered are:

1. doorways - Because radioactive sources are usually integrated with large, heavy shielding, motorized carts, trucks, or fork lifts may be needed to move them. Doorways and hallways leading to exits should be large enough to allow passage of these machines.
2. ramps - Sealed, shielded radioactive sources can weigh tons and may exceed the lifting capacity of freight elevators. Gently sloping ramps should be

provided between floor levels so that such sources can be transported by fork lifts from one building level to another.

3. ceiling openings - Ceilings can be designed so that the roof is easily dismantled, providing an opening large enough for a crane to lift out a shielded source or any large heavy object.

Section 8.5 CONTROL OF EXTERNAL RADIATION

Dose rates to personnel from radioactive materials can be greatly reduced by the placement of attenuating or shielding materials between personnel and the radiation source. The shielding materials can be designed into the building structure or they can be separate from the building. Shielding may be required to protect personnel from radiation emitted from open, unsealed radioactive materials and from radiation-generating devices. External dose rates are also controlled by restricting access to radiation areas through the use of interlocks, warning systems, and guards. (See Chapter 6 for details on the control and reduction of external exposure.)

8.5.1 Shielding Requirements

Shielding is required wherever the anticipated dose-equivalent rate will exceed 2.0 mrem/hr. The shielding should reduce the dose-equivalent rate to 0.2 mrem/hr or less.

A. Integrity. Shielding must be designed so that the degree of protection is constant from all angles of approach. The simplest method of achieving uniform protection is to surround a source with a uniform shield. In practice, however, a shield is usually penetrated by cooling pipes, electrical power and signal cables, rotating shafts, and removable plugs or covers, and special considerations must be made for these penetrations in the shield. Design features such as shadow shields, baffles, and offsets can help ensure adequate protection.

B. Materials. The choice of shielding material depends upon factors such as cost and the desired thickness and mass of the shield. However, all of the following should be considered whenever shielding material is being selected:

1. attenuation characteristics - Different shielding materials have different abilities to attenuate photons, neutrons, and beta particles.
2. structural integrity - The material selected must be structurally stable.
3. nonflammability - The shielding material should be fire resistant or noncombustible and should not release toxic gases or smoke when heated.
4. confinement capability - The shielding material may have to contain gases, solids, and liquids in the shielded enclosure.

Shielding materials commonly used for various types of radiation are described below.

(1) Shielding for Ions and Electrons. Virtually any material can be used as shielding for ion and electron sources as long as the shield is thicker than the range of the particles. Bremsstrahlung radiation may be produced if shielding materials with a high Z number (atomic number), such as lead or iron, are used. The likelihood of bremsstrahlung radiation can be reduced by the use of low-Z shielding materials, such as plastics. If bremsstrahlung radiation is produced, it can be attenuated by lead, iron, or any material that shields against x and gamma rays (see Chapter 1).

(2) Shielding for X- and Gamma-Ray Sources. Common shielding materials for photon sources are lead and iron. Depleted uranium and tungsten are expensive materials for shielding but they can be used if a relatively thin shield is required. Concrete and water can be used if the thickness of the shield is of no consequence.

(3) Shielding for Neutrons. Shielding for thermal (slow, or low-energy) neutrons is provided by thin layers of materials that have a high cross section for capture, for example, boron or cadmium. A disadvantage of cadmium is that, after a neutron is captured, the material emits high-energy gamma rays for which shielding must also be provided.

Fast neutrons are not easily shielded. In addition, sources of fast neutrons are commonly also sources of gamma rays; the shielding material must therefore be able to shield against both the photons and the neutrons. Shielding of fast neutrons is generally a two-step process. First, hydrogen-

containing materials such as plastic, water, or concrete are used to moderate, or reduce, the neutron energies to thermal levels by elastic scatter. If the shielding material contains high-Z elements, such as lead or iron, then the neutrons may lose their energy through inelastic collisions. The thermal neutrons may then be captured, as described above, by boron, cadmium, or (to a lesser extent) the hydrogen in water.

C. Entryways. Wherever possible, entryways should consist of a labyrinth, or passage with turnings, that scatters radiation twice before it hits a door. This scattering reduces the amount of radiation reaching the door, with two positive results: first, the likelihood of radiation streaming around the door is lowered; and second, the shielding requirements for the door are reduced and the weight of the door is thus lowered. Labyrinths can reduce the shielding requirements for a door to negligible levels.

D. Quality Assurance. Following the construction of any shield, the shield must be tested for uniformity. In concrete, for example, voids may occur or the aggregate may settle, making the shielding characteristics uneven and unacceptable. Special scrutiny should be given to all penetrations and to the crevices between concrete blocks, if they are used.

8.5.2 Access Restrictions for Radiation Areas

Access to radiation areas should be restricted whenever the dose-equivalent rate exceeds the levels that define a radiation area (see Section 8.4.2), and shall be restricted whenever the dose-equivalent rate exceeds the level that defines a high-radiation area. Requirements for access restrictions are defined in 10 CFR 20.203. Access may be restricted by interlocks and warning systems or by guards.

A. Interlocks and Warning Systems. An interlock is an electromechanical device such as a switch that causes a radiation-generating device to stop producing radiation if the access barrier to the device is violated. Examples of interlocks include:

1. door interlocks - These interlocks turn off the radiation-generating device if the door to the high-radiation area is opened; they also prevent operation of the device until the door is closed.

2. device-mounted switches - A switch integrated with a timer is mounted on the radiation-generating device. To operate the device, the operator must enter the room, turn on the switch (and timer), leave the room, and close the door before starting the device from outside the room. The purpose of this type of switch is to ensure that the operator enters the room before every procedure is begun and instructs all personnel to leave the room. The timer allows sufficient time for all these steps to be performed without rushing.

3. emergency shutoff or SCRAM switches - These switches are located throughout the room containing the radiation-generating device. Their purpose is to allow personnel inadvertently left in the high-radiation area to shut off the device or prevent it from starting up. These switches must be reset before the device can be operated.

Warning systems may consist of lights or alarms or both, as follows:

1. lights - Rotating red warning lights (the kind used on emergency vehicles) are located near eye level and are bright enough to be seen anywhere in the exposure room even if not viewed directly. The lights should be on for 15 seconds before an irradiation starts and during the entire irradiation.

2. alarms - Warning alarms sound for 15 seconds before an irradiation can start. When irradiation is started after the 15-second delay, lights remain on and audible alarms stop.

All interlocks and alarm systems shall be fail-safe so that a radiation-generating device cannot be operated if the warning systems or interlocks are inoperable. Signs describing the systems and how they are used should be posted near each interlock or warning system.

B. Guards. Security guards can prevent unauthorized personnel from entering radiation areas by checking the credentials of each individual who desires entry. Security guards are necessary when electrical or mechanical devices for restricting access have been inactivated (for repair or testing), the radioactive material is at a temporary location, or national security requires the use of guards.

Section 8.6 CONTROL OF INTERNAL RADIATION

Internal radiation is controlled by the use of 1) containment devices, which prevent radioactive materials from entering work areas where they might be inhaled or ingested by personnel; 2) ventilation systems, which remove radioactive materials from the air of work areas to ensure clean breathing air; and 3) air-sampling and air-monitoring systems, which have alarms to notify personnel if concentrations of radioactive materials exceed permissible limits. (See Chapter 5 for details on the control of internal exposure.)

8.6.1 Containment Devices

The spread of radioactive materials can be kept to a minimum by the use of sealed sources and containment devices such as hoods, glove boxes, and hot cells.

A. Sealed Sources. A sealed (or encapsulated) source is defined as a radioactive source sealed in a container that has a banded cover. The containers are designed not to rupture and thus to prevent dispersion of the radioactive material under normal operating conditions and following minor accidents, such as a container inadvertently being dropped. The integrity of sealed sources should be tested as described in Chapter 4.

B. Hoods. Open-face or fume hoods should be designed and located to provide constant air flow into the hood. The velocity of the air flowing into the hood (the face velocity) must be sufficient to ensure that no contamination enters the room. For conventional hoods, a face velocity of 46 ± 8 linear meters/min meets this criterion. Supplied-air hoods and National Cancer Institute hoods have other criteria; consult the manufacturer's literature for details concerning a specific hood.

Hoods should be illuminated with lights that can be serviced from outside the hood. Outlets for gas, air, and water should be located along the back or sides of the hood and should be controlled through knobs located outside the hood. Electrical outlets should be on the outside of the hood.

Each hood should be strong enough to support all necessary shielding, which should attenuate radiation in all directions. The air from each hood

should be exhausted through HEPA filters of a type appropriate for the laboratory classification (see Section 8.4.3). In all cases, a prefilter should be placed ahead of the HEPA filters. The exhaust ducts should be designed to allow in-place testing of the filter systems. In addition, pressure taps should be provided to allow measurement of the pressure drop across the filters. The filters should be located to allow rapid, clean servicing with little danger of the workplace being contaminated.

The following general rules have been established for the design of hoods for work with radioactive and chemically toxic materials (Industrial Ventilation 1980); they are applicable for glove boxes and hot cells as well.

1. Operations in which radioactive materials are handled should, as often as possible, be performed in enclosed areas to prevent the contamination of large air volumes.
2. High-velocity cross-drafts should be avoided because they may increase contamination and dust loading.
3. The volume of air withdrawn from the hood must be larger than the volume of contaminated gases, fumes, or dusts created in the hood.
4. If possible, operations requiring large amounts of wet digestion, volatilized acid, or solvent treatment should be confined to one group of hoods, and dry material should be handled in others.
5. Whenever possible, radioactive aerosols should be removed by filtration. The filters should be as close to the hood as practical to prevent unnecessary contamination of equipment and ductwork.
6. The value or accountability of the material used in a hood may require that the hood be designed so that even the smallest chips and turnings can be reclaimed.
7. A supply of coolant inside the hood may be needed, depending on the pyrophoric nature of the contaminant (its ability to ignite spontaneously).
8. Hoods and duct systems should be designed to be easily accessible for decontamination, and should be constructed of materials that are easily decontaminated. For this reason, stainless steel is frequently used for the metal parts of hoods.

9. The hood fan should be located close to the release point from the building so that ductwork within the building is under a negative pressure.

C. Glove Boxes. Glove boxes can be designed to function as a primary containment to minimize the potential for release of radioactive materials. The use of glove boxes minimizes contaminated-air volumes and simplifies air treatment problems. Glove boxes should be designed to operate at a negative pressure (1.8 ± 0.64 cm water gauge pressure) with respect to the room in which they are located. They should be equipped with differential gauges to measure the pressure drop and with control devices to prevent excessive vacuum or pressure build-up. Penetrations in the glove box (e.g., conduits, ports, ducts, and windows) should be sealed to prevent the release of radioactive materials.

D. Hot Cells. Hot cells are specialized rooms in which large quantities of radioactive materials are used. The cells are normally fitted with remote manipulators, which allow the manipulation of nuclides that emit gamma rays and high-energy beta particles without personnel receiving excessive radiation doses to the hands, wrists, and forearms. Hot cells are maintained under negative pressure to minimize the spread of radioactivity in the event of a leak. The exhaust should be filtered through two HEPA and charcoal filters.

8.6.2 Ventilation Systems

Ventilation systems are an essential part of a building's safety features. Consequently, they should be designed to complement the building layout and should remain functional or fail-safe during all operations and all credible accidents.

The ventilation system must confine airborne radioactive materials within the appropriate areas of the building. It should be capable of removing from routinely occupied areas any airborne radioactive materials resulting from normal or accident conditions. Further, the ventilation system should be designed to clear all normal or accidentally generated effluents from the air before the air is released to the environment.

A. Ventilation Zones. The ventilation system should include physically separate ventilation zones to prevent cross-contamination of air. Ventilation Zone I should correspond to the confinement portion of the radiation area (i.e., hoods, glove boxes, and hot cells). Ventilation Zone II should correspond to the remainder of the radiation area and to controlled areas. Ventilation Zone III should correspond to uncontrolled areas. Ventilation Zone III is required for buildings containing predominantly Class A laboratories that need office support. (Ordinarily, Class A laboratories should be in separate buildings with minimal office space; in these areas, ventilation Zone III is optional for the uncontrolled areas.)

B. Air Flow Patterns. Air should flow from the ceiling to the floor of a laboratory and should not flow directly across bench tops. In general, laboratories should be designed to provide draft-free conditions to keep the movement of particulate matter by air currents as low as possible.

The air flow for the whole building and for individual laboratories should be from areas of low (or no) radioactivity to areas of progressively higher activity. This direction of flow ensures that material that may become airborne will not contaminate other areas in excess of their permitted limits.

C. Pressure Differentials. Pressure differentials should be used to maintain the desired air flow characteristics. The exhaust system should be used to keep areas with relatively high activity levels at a negative pressure relative to the rest of the building. The building itself should have a negative pressure relative to the outside. In order to maintain the proper pressure differentials and keep the air flowing in the desired direction, the supply fan delivering air to laboratories should be controlled by interlocks that automatically shut off the air supply so that it is impossible to deliver air to the laboratories when the exhaust system is shut down for any reason.

D. Duct Routing. Exhaust ducts in multistory buildings should be routed to common ducts, or plenums, that are easily accessible. In addition, ducts should be labeled as to their point of origin. For single-story buildings, hoods should be vented to the roof using the least possible amount of ducting inside the building.

E. Filtration

(1) Type and Location. Filters or traps for exhaust air are required to ensure that release levels are kept ALARA. Filter systems should be designed for easy access, removal, contamination control, and in-place testing. In general, exhaust filters should be placed close to the hoods, glove boxes, and hot cells in order to avoid contaminating ventilation duct systems.

(2) Backflow Prevention. If the air flow through a filter were reversed, radioactive particulates could be pulled into a laboratory, with serious consequences. For this reason, filters that routinely become burdened with radioactive particulates should be protected by dampers that restrict the reverse flow of air. Inverse-flow dampers can be simple, weighted, shutter-like dampers that open passively with positive air flow. In dampers with more complex designs, electrical mechanisms keep the dampers open, and springs or pressurized air ensures their closure if the electrical supply is disrupted.

(3) Testing. Filters should be designed so that they can be tested in place. A DOP mist is used to test HEPA filters. Charcoal filters can be tested using a gaseous halogenated-hydrocarbon refrigerant, in accordance with Section 12 of the American National Standards Institute's (ANSI) Standard N510-1975, to ensure that bypass leakage through the absorber section is less than 0.05%.

(4) Maintenance Accessibility. Ventilation filters and blowers require periodic removal and replacement. Filter systems are often contaminated at the time of their replacement, and maintenance personnel must be protected against possible inhalation of radioactive dusts, mists, and fumes during filter replacement. External exposure is also a potential problem if the filters are loaded with radionuclides that emit gamma rays or high-energy beta particles. The filter units should be placed so that individual filters can be removed easily without the need for scaffolding. If scaffolding is required, however, enough free floor space should be available for the installation of the scaffolding.

8.6.3 Sampling and Monitoring Equipment

An important aspect of facility design is to provide for sampling and measurement of the concentration of airborne radioactive materials and for monitoring of the radiation levels in the workplace. As discussed in Chapter 5, sampling is the collection of air that is then analyzed for activity levels at a later time and in a different place; monitoring, on the other hand, is the continuous reading of the radiation level in a facility by a radiation detection instrument. Types of sampling and monitoring equipment are discussed in Chapter 2.

All sampling and monitoring instruments should have lights that indicate whether the instrument is turned on, in standby mode, or not operating. These lights, or status indicators, should be readily visible from any work area. All monitors should be provided with both visual and audible alarms. The instruments should be designed so that, if an alarm has been tripped, the instrument must be rest manually; automatic cessation of the alarm function is not acceptable.

A. Air Samplers and Monitors. All Class A and B laboratories should be equipped with fixed systems for sampling and monitoring the air. The sampling heads should be placed where releases could occur, as well as in front of each room's air exhaust.

Areas occupied by personnel where concentrations of airborne radionuclides may exceed the concentrations given in 10 CFR 20, Appendix B, Table I, should contain a continuous-monitoring device that activates an alarm when the airborne concentration exceeds 25% of the values given in the table.

B. Radiation Area Monitors. Continuously operating area monitors should be provided to measure the ambient dose-equivalent rate wherever that rate may exceed 50 mrem/hr. An alarm on each radiation monitor should notify workers if the device is not operating. Each radiation monitor should actuate audible and visual alarms whenever a preset radiation limit has been exceeded. The instruments should be capable of measuring dose-equivalent rates in the range of 10,000 mrem/hr. Finally, the detector portions of the monitors should be easily replaced and should be located where they can be calibrated in place.

Section 8.7 FACILITY SUPPORT

Other considerations in facility design are the provision of change rooms, decontamination facilities, and separate supply and sewer systems for sanitary and process water.

8.7.1 Change Room Facilities

Rooms in which workers can change clothing should be available and should be designed to prevent cross-contamination. Each worker should have two lockers, one for clean clothing and another for potentially contaminated clothing. Showers should also be provided in the change rooms. Change rooms may be separate from or part of personnel decontamination facilities.

8.7.2 Personnel and Property Decontamination Facilities

Facilities for the decontamination of personnel and property should be available. Decontamination facilities for personnel should have showers. The shower drains should be separate from the sanitary sewer system and should empty into a holding tank if contamination levels are expected to be high.

Facilities for the decontamination of property should be large enough to accommodate the largest piece of equipment. Each facility should include a hood and should have drains that are directed to holding tanks.

8.7.3 Water Supply and Sanitary Sewers

Sanitary water provided in radiation areas shall be used for safety showers and fire protection sprinklers only. Drinking fountains should not be located in radiation areas. Process water supplied to radiation areas shall be isolated from sanitary water systems by the use of either separate systems or back-flow preventors.

Sinks in radiation areas should not be equipped with drains connected to a sanitary sewer. If sinks and drain lines are connected to a sanitary sewer, they shall be so labeled, and the discharge of radioactive wastes to any sanitary sewer shall be prohibited.

REFERENCES

American National Standards Institute (ANSI). 1975. Testing of Nuclear Air-Cleaning Systems. ANSI N510, American Society of Mechanical Engineers, New York.

Fitzgerald, J. J. 1969. Applied Radiation Protection and Control, Volume 1. Gordon and Breach, New York.

Industrial Ventilation - A Manual of Recommended Practice. 1980. American Conference of Governmental Industrial Hygienists. Available from the Committee on Industrial Ventilation, P.O. Box 16153, Lansing, Michigan 48901.

Kathren, R. L., and J. M. Selby. 1980. A Guide to Reducing Radiation Exposure to As Low As Reasonably Achievable (ALARA). DOE/EV/1830-T5, National Technical Information Service, Springfield, Virginia.

National Council on Radiation Protection and Measurements (NCRP). 1978. Operational Radiation Safety Program. NCRP 59, Washington, D.C.

U.S. Code of Federal Regulations. 1982. Title 10, Part 20, "Standards for Protection Against Radiation." U.S. Government Printing Office, Washington, D.C.

U.S. Department of the Army, Headquarters. Safety - Ionizing Radiation Protection (Licensing, Control, Transportation, Disposal, and Radiation Safety). AR 385-11, Washington, D.C.

U.S. Department of the Army, Headquarters, Army Materiel Command. Safety - Radiation Protection. DARCOM-R 385-25, Washington, D.C.

CHAPTER 9. TRANSPORTATION OF RADIOACTIVE MATERIALS

FIGURES

TABLES

CHAPTER 9. TRANSPORTATION OF RADIOACTIVE MATERIALS

Radioactive materials are one of 13 classes of hazardous materials for which the Department of Transportation (DOT) regulates shipping. (Examples of other hazardous materials include explosives, poisons, flammable liquids, and corrosive materials.) The purpose of this chapter is to provide guidance to radiation protection or transportation personnel responsible for both onsite and offsite shipments of radioactive materials. The chapter includes definitions of some terms frequently used in shipping. Requirements for packaging, shipping, and receiving certain types of radioactive material are discussed, and information on specific packaging procedures and containers for selected types of radioactive materials is provided in the appendixes. For additional regulations that relate to the shipment of radioactive wastes, see Chapter 10.

The shipment of 15 grams or more of fissile material requires special packaging procedures and is beyond the scope of this manual. Fissile material includes ^{238}Pu, ^{239}Pu, ^{241}Pu, ^{233}U, ^{235}U, or any articles containing these radionuclides. For assistance with this type of shipment, contact the installation office responsible for transportation. If hazardous materials other than radioactive materials must be shipped, contact Headquarters, DARCOM, Materiel Management for assistance.

Section 9.1 REGULATING AGENCIES

The shipment of radioactive material between states by rail, air, road, or water is regulated by DOT as specified in Title 49, "Transportation," of the Code of Federal Regulations, Parts 100-199 (49 CFR 100-199). Postal shipments are under the jurisdiction of the U.S. Postal Service and must comply with the postal regulations in the U.S. Postal Manual. In special cases, as will be discussed later, the U.S. Nuclear Regulatory Commission (NRC) also regulates the shipment of radioactive material, as specified in 10 CFR 71. Information concerning the transportation of radioactive material can also be found in AR 385-11 and MIL-STD-1458.

The shipment of radioactive material requires careful planning. The shipper must be familiar with all applicable shipping regulations. This chapter is intended only to provide guidance and to serve as a reference to the shipper of radioactive material. Users of this manual should obtain the latest copy of 49 CFR and review its transportation regulations before making a shipment.

Section 9.2 TERMINOLOGY

Personnel responsible for shipments of radioactive materials need to be familiar with the terminology used in DOT regulations.

9.2.1 General Terms

A. Radioactive Material. Radioactive material is any material or combination of materials that spontaneously emits ionizing radiation (see Chapter 1). A shipment of articles that contains more than 0.002 μCi/g of radioactive material is regulated by DOT.

B. Transport Group. Seven transport groups are used to classify radionuclides according to their radiotoxicity and their relative potential hazard in transportation. Transport Group I includes those materials considered the most radiotoxic, for example, plutonium, americium, and radium, all of which emit alpha particles. Radionuclides assigned to Transport Group VII are considered the least radiotoxic. The transport group is one of several factors that affect how much of a given radionuclide can be transported; the total activity level permitted in a package of Transport Group I material is lower than the activity level permitted in a package of Transport Group VII material.

A list of radionuclides and the transport groups in which they belong is presented in Appendix A. If a radionuclide is not listed in Appendix A, it is assigned to a transport group according to Table 9.1. For a material or mixture of materials (such as radioactive waste) that contains more than one radionuclide, the following rules apply:

TABLE 9.1. Transport Group Classifications for Radionuclides Not Listed in Appendix A

Radionuclide	Radioactive Half-Life: 0 to 1000 Days	1000 Days to 10^6 Years	Over 10^6 Years
Atomic Number 1-81	Group III	Group II	Group III
Atomic Number 82 and over	Group I	Group I	

(a) 49 CFR 173.390 (1980).
(b) No unlisted radionuclides can be assigned to Group IV, V, VI, or VII.

1. If the identity, transport group, and activity of each radionuclide are known, the sum of the ratios of total activity to permissible activity for each transport group must not exceed 1.

2. If the transport groups of the component radionuclides are known but the amount or total activity in each group cannot reasonably be determined, the mixture is assigned to the most restrictive group represented among the components.

3. If the identity of all or some of the radionuclides cannot be reasonably determined, each of the unidentified radionuclides is considered as belonging to the most restrictive group (Group I).

4. Mixtures consisting of a single radioactive decay chain with the radionuclides in naturally occurring proportions are considered to consist of a single radionuclide. The transport group and activity are considered to be those of the first member in the chain, unless a daughter radionuclide has a half-life longer than that of the first member and an activity greater than that of any other member of the chain at any time during transportation; in that case, the transport group of the daughter nuclide is used, and the activity of the mixture during transportation is considered to be the maximum activity of that daughter nuclide.

C. Transport Index. The transport index is the radiation dose-equivalent rate, in mrem/hr, at 1 m from any accessible package surface. The transport index is placed on shipping papers and on the package label to provide an

indication to the shipper of the degree of control required for the package during its transport. For example, the transport index is used to decide how many packages can be shipped in one load and where the packages should be placed on the truck.

The transport index is defined differently for shipments of fissile material. Information about the transport index for fissile material can be found in 10 CFR 71 and 49 CFR 173.396.

D. Special-Form Radioactive Material. Special-form radioactive material is defined as material that, if released from a package, may present a hazard because of direct external radiation, but that has little possibility of contaminating any object or person it comes in contact with because of its high physical integrity. To qualify as special-form material, an item must either be a massive, solid object made of a metal or alloy, or be encapsulated. The dimensions of the special-form material must all be larger than 0.5 mm, or the item must have at least one dimension larger than 5 mm.

Department of Transportation regulations specify rigid performance tests for special-form material. These tests are used to ensure that special-form material will maintain its integrity and not scatter radioactive material to the environment if its package fails during shipment. Two examples of performance tests are the percussion test and the free-drop test. When subjected to these tests, massive solid forms must not break, crumble, or shatter, and capsules containing radioactive material must keep all of their contents. To perform the percussion test, a lead sheet on a flat, unyielding surface is needed. The capsule or material is placed on the lead sheet and must maintain its integrity when the flat circular end of a steel rod with a 2.5-cm diameter is dropped on it from a height of 1 m. The free-drop test requires that the capsule or material be dropped 9.1 m onto a flat, unyielding surface without loss of contents. Other performance tests specified by DOT include a heat test and an immersion test. For more information concerning performance tests for special-form material, refer to 49 CFR 173.398. Because special equipment may be needed to conduct these tests, the radiation protection officer (RPO) or transportation officer should obtain qualified assistance.

E. Normal-Form Radioactive Material. Normal-form radioactive material is material that does not qualify as special-form material. Examples of normal-form material include radioactive powder in a glass or plastic bottle, radioactive waste in a plastic bag, and radioactive liquid in a metal container.

F. Specification Packaging. Specification packages are those specified by package design and use in 49 CFR 173 and 178. Packages used to ship radioactive material must be designed and constructed so that radioactive material will not be released to the environment at any time during the shipment. Package specifications vary according to the type of material being shipped. Details concerning package specifications for various types of radioactive material are considered in later sections of this chapter.

G. Specification Marking. Each specification package described in 49 CFR is assigned an identifying mark consisting of letters and numerals. For example, the specification marking assigned to a plywood shipping box would be DOT-7A (48 CFR 178.350). The specification marking must be placed on an unobstructed area of the container. The name and address or symbol of the installation, unit, or firm making the package mark must be included. The color used for the letters and numerals must contrast with the package. The markings must be at least 1.3 cm high and must be permanently applied to the package by stamping, embossing, burning, or printing.

H. Specification Labeling. Specification labels are those specified by design and use in 49 CFR 173. They vary according to the material shipped and are described later in this chapter.

9.2.2 Terms Used to Define Quantities of Radioactive Materials

A special group of terms is used in DOT regulations to classify quantities of radioactive materials according to their activity, form, and transport group. These terms are limited quantity, low specific activity, Type A quantity, Type B quantity, and large quantity. Given a package of radioactive material of a particular form and transport group, the total activity or specific activity of the packaged material determines the quantity classification, which in turn determines the packaging, labeling, and handling requirements that must be met for that package.

A. Limited Quantity. If a package of radioactive material in normal form, or a manufactured article or device that contains radioactive material (other than liquid) in a nondispersible form, has an activity level that does not exceed that listed in Columns 2, 3, or 4 of Table 9.2 for the appropriate transport group, then the package can be shipped as a limited quantity of radioactive material (49 CFR 173.391(b)). A package of special-form material can be shipped as a limited-quantity package if its activity does not exceed 1 mCi. A package shipped as a limited quantity may not contain more than 15 grams of fissile material.

Two items in Table 9.2 should be noted. First, the amount of material (the permissible activity level per package) that can be shipped in the limited-quantity category increases with increasing transport group number. (Remember that assignment to a transport group is dependent on the radiotoxicity of the material, with Transport Group I containing the most radiotoxic material.) Second, the activity limit for shipping special-form material as a limited-quantity package is independent of the transport group.

Manufactured articles (other than reactor fuel elements) in which the radioactive material is metallic natural or depleted uranium, natural thorium, or alloys of uranium or thorium can be shipped as limited-quantity packages. Tritium oxide in aqueous solution with a concentration less than 0.5 mCi/ml and a total activity per package of less than 3 Ci is also considered a limited quantity.

B. Low Specific Activity. A package of radioactive material with a concentration that does not exceed that listed in Table 9.3 for the appropriate transport group can be shipped as a low-specific-activity (LSA) package. A wide variety of commodities can be considered for LSA shipment. For example, this class could include residue or solutions from chemical processing; waste such as building rubble; wood and fabric scrap; and metal, glassware, paper, cardboard, sludge, and ash. Low-specific-activity material may also include unirradiated natural and depleted uranium, unirradiated natural thorium, uranium or thorium ores, and tritium oxide in aqueous solution, provided that the concentration does not exceed 5 mCi/ml.

TABLE 9.2. Quantity Classifications for Radioactive Materials (49 CFR 173)

Transport Group	Maximum Permissible Activity Per Package (Ci) for Each Classification					
	Limited Quantity			Type A Quantity	Type B Quantity	Large Quantity
	Radioactive Material(a)	Radioactive Device(b)				
I	0.00001	0.0001(c)	0.001(d)	0.001	20.0	20.0
II	0.0001	0.001	0.05	0.05	20.0	20.0
III	0.001	0.01	3.0	3.0	200.0	200.0
IV	0.001	0.05	3.0	20.0	200.0	200.0
V and VI	0.001	1.0	1.0	20.0	5,000.0	5,000.0
VII	25.0	25.0	200.0	1,000.0	50,000.0	50,000.0
Special-Form Material	0.001(e)	0.05	20.0	20.0(f)	5,000.0	5,000.0

(a) Includes tritium oxide in aqueous solution with a concentration not exceeding 0.5 mCi/ml and with a total activity per package of not more than 3.0 Ci.
(b) Radioactive devices include manufactured articles such as instrument clocks, electronic tubes, or equipment containing limited quantities of radioactive material (no liquids) in a nondispersible form. The radiation dose rate 10 cm from an unpackaged device may not exceed 10 mrem/hr and the dose rate at the external surface of the package may not exceed 0.5 mrem/hr.
(c) Numbers in this column represent activity per device.
(d) Numbers in this column represent activity per package.
(e) Limited to 15.0 grams of fissile material.
(f) ^{252}Cf is limited to 2.0 Ci.

TABLE 9.3. Concentration Limits for Low-Specific-Activity Packages (49 CFR 173)

Transport Group	Concentration Limit (mCi/g)
I	0.001
II	0.005
III	0.3
IV	0.3

Nonradioactive articles that have been externally contaminated with radioactive material can be shipped as LSA material provided that two conditions are met. First, the radioactive material must not be readily dispersible. Second, the surface contamination, when averaged over an area of 1 m^2, must not exceed 0.0001 mCi/cm^2 for Transport Group I radionuclides or 0.001 mCi/cm^2 for other radionuclides.

C. Type A, Type B, and Large Quantity. The activity limits that determine whether a package of radioactive material is classified as a Type A quantity, Type B quantity, or large quantity are presented in Table 9.2. The maximum permissible activities for packages in these three categories are higher than the maximum permissible activities for packages in the limited-quantity and LSA categories. Note once again that special-form material is not classified by transport group.

Section 9.3 SHIPPING REQUIREMENTS AND PROCEDURES - OFFSITE

The regulations and requirements that must be met for shipping radioactive material offsite depend on the nature and quantity of the material to be shipped. The flow chart in Figure 9.1, together with the definitions in Section 9.2, can be used to characterize the material. General packaging requirements are described below, followed by specific requirements for the different classes of packages.

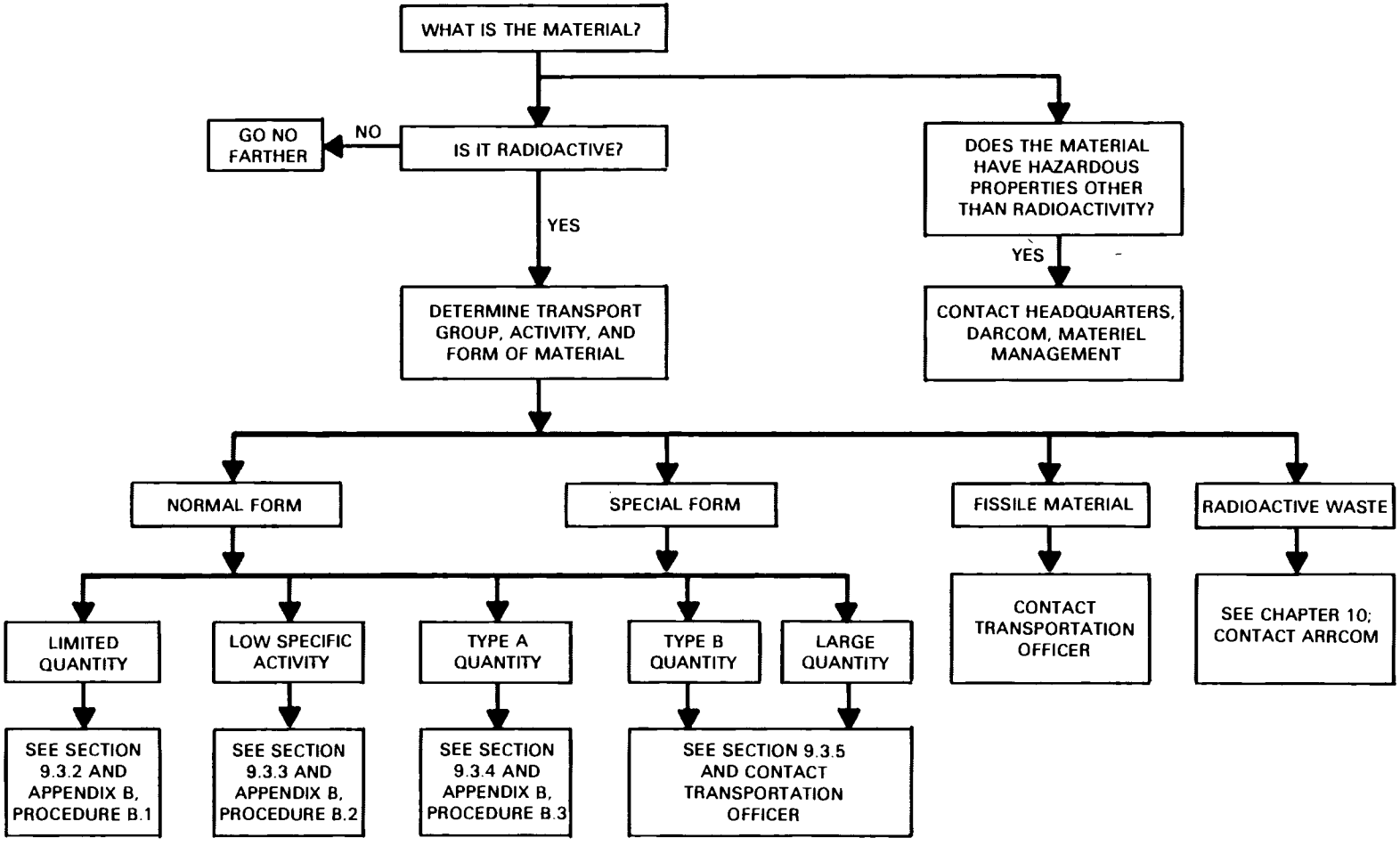


FIGURE 9.1. Material Characterization and Packaging Requirements

9.3.1 General Packaging Requirements and Procedures

Some requirements apply to all packages of radioactive materials (49 CFR 173.24, 173.393). In general, packaging for radioactive materials must be leaktight and must protect the internal contents of the package. The package must be designed so that the radioactive material is not released from the package under normal transportation conditions. Persons responsible for packaging radioactive material should remember that the package could be dropped, kicked, punctured, or thrown, or have heavy packages thrown on top of it. Each package must be strong, tight, and rugged.

A. Packaging Materials. High-quality materials capable of withstanding any abuse during transportation should be used to package radioactive materials. For example, lumber must be well-seasoned, dry, and free of defects (such as loose knots) that would lessen the strength of the package. Steel must be low-carbon, commercial-quality steel. Examples of acceptable steel include stainless, open-hearth, electric, and basic oxygen steel. Polyethylene must not be permeable to liquids or gases.

B. Package Construction. Good construction techniques are essential for shipping containers. Nails and staples should not puncture the contents of the package.

C. Multiple Packaging. Multiple packaging should be considered for radioactive material. Multiple packaging means that the item to be shipped is placed inside a container, which is then placed inside another package. Cushioning or internal bracing is used to ensure that the radioactive material does not move around inside its package during shipping. This provides further protection for the radioactive material and ensures that the external dose-equivalent rate of the package remains constant.

D. Dose-Equivalent Rate at the External Surface. All radioactive material must be packaged so that 1) the dose-equivalent rate at any point on the external surface of the package does not exceed 200 mrem/hr, and 2) the dose-equivalent rate 1 m from any external surface does not exceed 10 mrem/hr (i.e., the maximum permitted transport index is 10). It may be necessary to shield the radioactive material within the package in order to achieve this

dose-equivalent rate at the package surface. Lead containers, sometimes referred to as lead pigs, are often used.

If the radioactive material is to be shipped in a closed transport vehicle that is being used exclusively for that purpose (an exclusive-use vehicle), the dose-equivalent rate at the surface of the package may be higher than the limit given above. However, the dose-equivalent rates at the vehicle surface and to individuals within the vehicle must be considered. For exclusive-use shipments, the following rules apply:

1. The dose-equivalent rate 1 m from the package surface may not exceed 1000 mrem/hr.
2. The dose-equivalent rate at any point on the external surface of the closed transport vehicle may not exceed 200 mrem/hr.
3. The dose-equivalent rate at any point 2 m from the outer surface of the vehicle may not exceed 10 mrem/hr.
4. The dose-equivalent rate at any occupied position in the vehicle may not exceed 2 mrem/hr.

E. Liquid Radioactive Material. Liquid radioactive material must be packaged within a leak-resistant and corrosion-resistant primary container. The container must be adequate to prevent loss or dispersal of the contents if the package is subjected to a 10-m drop test (49 CFR 173.393(g)(1)). The primary container must then be placed in an inner container strong enough to prevent the loss or dispersal of the radioactive contents of the primary container. The packaging materials must include enough absorbent material to absorb at least twice the volume of the radioactive liquid contents, or else the inner container must be placed within a second leak- and corrosion-resistant inner container.

If shielding is used within the package to decrease the external dose-equivalent rate, the absorbent material is placed inside the radiation shield. However, the absorbent material may be placed outside the shield if the inner radioactive-material container and its shield are placed in a second leak- and corrosion-resistant inner container that could retain the radioactive contents of the first inner container if it broke during transport. The absorbent

material may also be located outside the shield if the dose-equivalent rate at the surface of the package would not be greater than 1000 mrem/hr, even if the radioactive liquid contents were taken up by the absorbent material.

F. Contamination Control. Every package must be monitored before it is shipped to ensure that there is no significant removable contamination on its surfaces. Removable contamination is considered significant if the level of contamination, when averaged over any 300-cm^2 area of the package, exceeds the maximum permissible levels shown in Table 9.4. When a package is being checked for surface contamination, a sufficient number of measurements must be taken to yield a representative assessment of the potential contamination level. Procedures for determining surface contamination levels are discussed in Chapter 4.

G. Package Dimensions. The smallest permissible dimension for any package containing radioactive material is 10 cm.

H. Package Seal. The outside of each package containing a Type A quantity, Type B quantity, or large quantity of radioactive material must have a seal that is not readily breakable and that, while intact, is evidence that the package has not been illegally opened.

TABLE 9.4. Removable-Contamination Limits for External Surfaces of Radioactive-Material Packages (49 CFR 173.397)

Contaminant	Maximum Permissible Limit	
	$\mu Ci/cm^2$	dpm/cm^2
Natural or depleted uranium and natural thorium:		
Beta-gamma	10^{-3}	2200
Alpha	10^{-4}	220
All other beta-gamma-emitting radionuclides	10^{-4}	220
All other alpha-emitting radionuclides	10^{-5}	22

9.3.2 Specific Requirements for Limited-Quantity Packages of Radioactive Material

Limited-quantity packages of radioactive material are exempt from specification packaging and labeling if the following conditions are met:

A. Packaging. The material must be packaged in strong, tight packages that prevent any leakage of radioactive material during normal conditions of transportation.

B. Dose-Equivalent Rate. The dose-equivalent rate at any point on the surface of the package must not exceed 0.5 mrem/hr. For manufactured articles, the dose-equivalent rate at 10 cm from any unpackaged device must not exceed 10 mrem/hr.

C. Contamination Level. There must be no significant removable surface contamination on the surface of the package (see Table 9.4).

D. Markings. The outside of the inner container must bear the marking "Radioactive Material."

E. Quantity of Radioactive Material. The total activity level of a package containing radioactive devices must not exceed the per-package limits shown in Table 9.2, Column 4. No package may contain more than 15 grams of fissile material. The total radioactive content of a manufactured article (except for reactor fuel elements) in which the only radioactive material is metallic natural or depleted uranium must not exceed 3 Ci, and the metallic uranium or thorium article must be enclosed in a nonradioactive, sealed metallic sheath (49 CFR 173.391(c)(4)).

9.3.3 Specific Requirements for Packages of Low-Specific-Activity Radioactive Material

Low-specific-activity radioactive material is also exempt from specification packaging, marking, and labeling when shipment is made in an exclusive-use vehicle and the following conditions are met:

A. Packaging. Material must be packed in strong, tight packages that prevent any leakage of material during transportation.

B. Contamination Level. The package exterior must not have any significant removable contamination.

C. Loading and Unloading. Shipments must not be transferred from one transport vehicle to another during shipment.

D. Liquid Radioactive Material. Liquid radioactive material must be packaged as described in Section 9.3.1(E).

E. Vehicle Condition. There must be no loose radioactive material in the vehicle. Shipments must be braced to prevent leakage or shifting of the load during transportation.

F. Placards. Except for shipments of unconcentrated uranium or thorium ores, the vehicle must be placarded.

G. Markings. The inner and outer containers must be stenciled or otherwise marked "Radioactive - LSA."

H. Shipping Instructions. Specific instructions for the maintenance of exclusive-use-shipment controls must be provided by the shipper to the carrier. These instructions must be included with the shipping papers.

9.3.4 Specific Requirements for Type A Packages of Radioactive Material

The packaging requirements for Type A quantities of radioactive material are much more stringent than those for limited quantities or LSA material. Specification marking, packaging, and labeling must be used. A Type A package must pass a series of tests to ensure that it can withstand rough handling. It must survive such test conditions as 1) being sprayed with water for 30 minutes and then, 1-1/2 to 2-1/2 hours later, being dropped from a height of 1.2 m onto an unyielding surface; 2) being dropped on each of its corners in succession from a height of 0.3 m; 3) having a 5.9-kg steel cylinder dropped on it from a height of 100 cm; and 4) being compressed by considerable weight for 24 hours. The package must be further tested to ensure that radioactive material is not released when the package is subjected to severe environmental conditions (e.g., extreme heat, cold, vibration, and pressure reduction).

More detail concerning standards and tests for Type A packages is given in 49 CFR 173.398. Contact the transportation officer for assistance in either testing containers or locating containers that meet the criteria for Type A shipments within the Department of the Army (DA) supply system.

9.3.5 Specific Requirements for Type B and Large-Quantity Packages of Radioactive Material

High-activity material is transported in packages that meet Type B and large-quantity criteria. These packages are intended to withstand severe accidents. They must meet all the requirements for Type A packages and pass an additional series of mechanical and fire tests. Containers for Type B and large-quantity shipments must be licensed. Contact the transportation officer for assistance.

9.3.6 Container Selection and Packaging Procedures

Container selection and packaging procedures are based on the quantity, kind, and form of the material being shipped, and on the mode and destination of the shipment. Specific packaging requirements for the various classes of packages have already been discussed. Packaging procedures for limited-quantity, LSA, and Type A quantity packages are detailed in Appendix B. Examples of specification containers are presented in Appendix C.

9.3.7 Warning Labels

Each package of radioactive material, except those containing exempt quantities or LSA material shipped under exclusive-use provisions, must be labeled on two opposite sides with one of three warning labels. These labels bear the unique trefoil symbol (Figure 9.2) and alert persons handling the package that it may require special handling. The labels are called radioactive white-I, radioactive yellow-II, and radioactive yellow-III. Which label is used depends on the dose-equivalent reading at the surface of the package and on the transport index for the package.

A. Radioactive White-I. The radioactive white-I label is used when the dose-equivalent rate at any point on the surface of the package is less than or equal to 0.5 mrem/hr.

FIGURE 9.2. Warning Labels for Packages Containing Radioactive Materials

B. Radioactive Yellow-II. The radioactive yellow-II label is used when the transport index will not exceed 1.0 during transport and the dose-equivalent rate at any point on the surface of the package is between 0.5 and 50 mrem/hr.

C. Radioactive Yellow-III. The radioactive yellow-III label is used when the dose-equivalent rate at the surface of the package is greater than 50 mrem/hr but less than or equal to 200 mrem/hr. The transport index may be greater than 1.0 but must be less than or equal to 10.0.

D. Material Exempt from Labeling. Labeling is not required when a package contains less than Type A quantities of radioactive material and the dose-equivalent rate at the surface of the package is less than or equal to 0.5 mrem/hr. However, in a limited-quantity package, the outside of the inner container must be marked "Radioactive." When an LSA package is being transported as an exclusive-use shipment, the outside of the package must be labeled "Radioactive - LSA."

E. "Empty" Labels. "Empty" labels (see Figure 9.3) must be attached to opposite sides of empty containers that formerly contained radioactive material. This label may be used only when there is no residual material or contamination that could cause the surface dose-equivalent rate to exceed 0.5 mrem/hr.

FIGURE 9.3. "Empty" Label for Empty Container

9.3.8 Placards

Radioactive-material placards, shown in Figure 9.4, must be attached to the four sides of a vehicle transporting packages of radioactive materials when any of the following is true:

1. the transport index is greater than 1.0
2. the shipment is a large-quantity shipment
3. the dose-equivalent rate at the surface of the shipping container exceeds 50 mrem/hr
4. the containers are marked "Radioactive - LSA."

9.3.9 Shipping Documents and Records

A. Consignee License. Unless exempted by the NRC, the consignee (or recipient) of a package containing radioactive material must have an NRC license to receive radioactive material. A DA permit is also required for a nonmilitary consignee (AR 385-11).

B. Bill of Lading. Each shipment of radioactive material must be accompanied by a bill of lading or delivery manifest. Any uniform bill of lading is acceptable.

FIGURE 9.4. Radioactive-Material Placard

C. Description of Material on Shipping Papers. A description of the radioactive material being shipped must be provided on the shipping papers. The information to be included is described in 49 CFR 172.201-204. The statement should include at least the following information:

1. name and address of the shipper and the consignee
2. the proper shipping name defined for the material in 49 CFR 172.101
3. the hazard class of the material as defined in 49 CFR 172.101
4. the name and mass number of each radionuclide in the shipment
5. a description of the physical and chemical form of the material if the material is not in special form
6. the activity of the material in each package of the shipment (stated in either curies, millicuries, or microcuries)
7. the type of DOT label applied (e.g., radioactive yellow-II)
8. the transport index assigned to each package that has a radioactive yellow-II or radioactive yellow-III label.

D. Shipper's Certification. The following statement must be signed by the shipper and be included on all shipping papers:

> "This is to certify that the above-named materials are properly classified, described, packaged, marked, and labeled, and are in proper condition for transportation according to the applicable regulations of the Department of Transportation."

E. Exclusive-Use Shipment. Specific instructions for the maintenance of exclusive-use-shipment controls must be provided to the carrier. These instructions must be included with the shipping papers.

F. Transportation by Air. If a package contains material prohibited for shipment by passenger aircraft, the words "cargo-only aircraft" must be included in the shipping papers after the description of the material.

G. Location of Papers During Shipment. The shipping papers that describe the radioactive material must be readily accessible for inspection or easily recognizable by authorities in the event of an accident (49 CFR 177.817(e)). The carrier should make the papers describing the hazardous material clearly distinguishable from other papers by tabbing them or by placing them in front of all other papers. The papers should be stored where they are within the immediate reach of the driver and readily visible to anyone entering the driver's compartment. They may be placed in a holder mounted on the inside of the door on the driver's side of the vehicle.

H. Records. Records of all shipments, including a copy of the bill of lading and monitoring records, should be retained for at least 5 years. Records demonstrating compliance with package design and performance standards should be retained for as long as the package design is in use and for at least 2 years after discontinuation of that design.

For shipments of special-form material, DOT requires that complete certification and a supporting safety analysis demonstrating that the material meets the required performance standards be kept on file for 1 year following shipment of the material. The RPO should be prepared to provide this information to DOT at any time during that year.

9.3.10 International Shipments

An agency of the United Nations, the International Atomic Energy Agency (IAEA), provides overall safety guidance for the international shipment of radioactive materials. The Inter-Governmental Maritime Consultative Organization (IMCO) and the International Air Transport Association (IATA) provide requirements for international shipments of radioactive material by sea and air, respectively (IMCO 1978, IATA 1981). The specific application and

enforcement of the regulations is the responsibility of each nation through which material is transported. Normally, a shipment that complies with the regulations of the nation of origin complies by agreement with the regulations of the nation through which the shipment is routed.

When ocean movement of radioactive material is planned, the shipping papers must include the following information:

1. proper shipping name (49 CFR 172.101)
2. classification (49 CFR 172.101)
3. pieces, weight, volume (49 CFR 172.202)
4. type of packaging (49 CFR 172.202)
5. name of radioactive material as listed in 49 CFR 173.390, 49 CFR 172.203
6. description of chemical and physical form (49 CFR 172.203)
7. specific activity (curies, millicuries, or microcuries) (49 CFR 172.203)
8. type of label (49 CFR 172.203)
9. transport index for each package bearing a radioactive yellow-II or yellow-III label (49 CFR 172.203)
10. "fissile exempt" 49 CFR 173.396, if applicable (49 CFR 172.203)
11. fissile class I, II, or III, if applicable (49 CFR 172.203)
12. DOT exemption, if applicable (49 CFR 172.203)
13. indicate "IMCO Class 7"
14. transport group
15. NRC license, if applicable
16. net weight of radioactive material
17. level of radiation at surface of package
18. level of radiation at 1 m from package
19. common commodity name of any item that contains radioactive material (e.g., radio tube, compass, electronic instrument, timepiece).

Section 9.4 SHIPPING REQUIREMENTS AND PROCEDURES - ONSITE

The movement of radioactive material within installation boundaries requires careful planning to prevent contamination of the installation environment. Military vehicles should be used, and the radioactive material must be loaded and transported according to the regulations in AR 55-355.

9.4.1 Packaging and Labeling

Sturdy containers free of removable surface contamination should be used for transporting radioactive materials onsite. The use of DOT-specification marking, packaging, and labeling is recommended but not necessary unless specifically required by an NRC radioactive-materials license.

9.4.2 Dose-Equivalent Rate

The dose-equivalent rate in any occupied area of the transport vehicle should be less than 2 mrem/hr. No one may receive a dose-equivalent of more than 100 mrem in any 7 consecutive days or 0.5 rem in any 1 calendar year. If the dose-equivalent rate in any occupied area of the vehicle exceeds 0.4 mrem/hr, film badges and radiation survey instruments must be used by personnel who accompany the shipment.

9.4.3 Supervision

When DOT-specification packaging, labeling, and marking are not used, the movement of radioactive materials within installation boundaries must be under the immediate supervision of the radiation protection personnel preparing the shipment. The shipment should be routed around areas in which explosives are stored or handled and areas where large numbers of people work.

9.4.4 Records

Records of all onsite shipments should be maintained for 5 years.

Section 9.5 RECEIVING

This section describes procedures to be used when shipments of radioactive material are received. Packages containing radioactive material and the vehicles used to transport them must be carefully monitored for excessive external radiation and contamination, to minimize the exposure of personnel and the spread of contamination. It is important to identify any personnel, vehicles, property, and facilities that have been exposed or contaminated as soon as possible so that remedial action can be taken.

9.5.1 Package Pickup

Packages that are not shipped directly to the receiver's installation must be picked up at the carrier's facility as soon as possible. When the carrier notifies the receiver of a package's arrival during the normal duty hours, the package should be picked up within 2 or 3 hours. If the carrier notifices the receiver after normal duty hours, the package may be picked up early the next work day. Packages that contain larger than Type A quantities of material must be delivered or picked up as soon as they arrive. When such shipments are expected, the carrier should be told who to notify if the shipment does not arrive during normal duty hours.

9.5.2 Monitoring Packages

Packages containing radioactive material should be monitored for excessive external radiation and contamination before they are unloaded from the transport vehicle at the receiving installation. If the package is not monitored at this time, the identification number of the transport vehicle should be noted by the receiver so that the carrier can be notified if contamination is found later. Packages should also be inspected for damage, and the integrity of any seal should be checked. The RPO must be notified immediately when damaged shipments are received. Requirements for reporting damaged shipments may be found in AR 385-11. If the control levels given below are exceeded, the vehicle should be isolated and measures taken to ensure that personnel who unload the shipment receive minimum exposure. Military personnel who may have been overexposed or contaminated will be

required by the commander to receive a medical examination at military facilities. Civilian personnel should also be encouraged to be examined at military facilities.

Packages that have been monitored in and removed from the transport vehicle should not be opened until they have been monitored again for contamination. This monitoring should be done within 3 hours of receipt of the package during normal duty hours, or within 18 hours if the package is received after normal duty hours. The receiver should become familiar with the contents of the package before opening it and should know the type and quantity of the radionuclide(s) in the package in order to determine what precautions should be taken in handling the material. The bill of lading is a useful reference.

Persons unpackaging radioactive material should wear disposable plastic gloves and work inside a radiation area. As the package is opened, the outer, inner, and primary containers should be monitored. For example, assume that a package containing a vial of ^{131}I is received. The vial has been packaged in a plastic bag, then in a lead container surrounded by absorbent packaging material, and then in a box. The box should be monitored first to be sure it is not contaminated. After the uncontaminated box is opened, the absorbent material should be checked for liquid that may have leaked from the vial. If the material is dry, the lead container should be monitored, then the plastic bag, and finally the vial to determine whether any contamination is present. Contaminated material should be decontaminated prior to storage. Remote-handling tools should be used for unpackaging material with high radiation levels.

A. Control Levels. Radiation levels must not exceed either 200 mrem/hr at any point on the package surface or 10 mrem/hr 1 m from the package surface. Removable contamination must not exceed 0.01 μCi per 100 cm^2 of package surface area monitored. If external radiation or radioactive contamination exceeds these limits, the delivering carrier should be notified immediately.

B. Procedure for Monitoring External Radiation Levels. A dose rate instrument should be used to measure the radiation level outside a package.

Measurements should be taken both 1 m from the package surface and as close to the package surface as possible without the instrument probe touching the package. If a preliminary survey of a package reveals radiation dose rates in excess of the control levels, the package should be moved away from other packages and resurveyed.

C. Procedure for Monitoring External Surface Contamination. Monitoring a package for external surface contamination requires two steps. First, a smear test is made on one or more random sections of the package surface by rubbing a filter paper over a predetermined area of the package surface (usually 100 cm^2). Second, the filter paper is taken to an area where the radiation level is at or near the background radiation level, and the activity on the paper is measured with a calibrated instrument. Information on selecting appropriate instruments for different types of radioactivity can be found in Chapter 2.

9.5.3 Monitoring Transport Vehicles

Vehicles used to transport radioactive materials must be monitored for radioactive contamination immdiately after the packages are unloaded. The procedure for this survey is described in Chapter 6. If contamination is found, the vehicle must be decontaminated before it is released.

9.5.4 Records

Records pertaining to the receipt of radioactive material should be maintained for 5 years.

REFERENCES

Inter-Governmental Maritime Consultative Organization (IMCO). 1978. Dangerous Goods Code. London, England.

International Air Transport Association (IATA). 1981. Restricted Articles Regulation. Quebec, Canada.

U.S. Code of Federal Regulations. 1982. Title 10, Part 71, "Packaging of Radioactive Materials for Transport and Transportation of Radioactive Materials Under Certain Conditions." U.S. Government Printing Office, Washington, D.C.

U.S. Code of Federal Regulations. 1980. Title 49, Part 172, "Hazardous Materials Table and Hazardous Materials Communications Regulations." U.S. Government Printing Office, Washington, D. C.

U.S. Code of Federal Regulations. 1980. Title 49, Part 173, "Shippers - General Requirements for Shipments and Packagings." U.S. Government Printing Office, Washington, D. C.

U.S. Code of Federal Regulations. 1980. Title 49, Part 177, "Carriage by Public Highway." U.S. Government Printing Office, Washington, D. C.

U.S. Code of Federal Regulations. 1980. Title 49, Part 178, "Shipping Container Specifications." U.S. Government Printing Office, Washington, D. C.

U.S. Department of Defense. Military Standard - Radioactive Materials: Marking and Labeling of Items, Packages, and Shipping Containers for Identification in Use, Storage, and Transportation. MIL-STD-1458, Washington, D.C.

U.S. Department of the Army, Headquarters. Military Traffic Management Regulation. AR 55-355, Washington, D.C.

U.S. Department of the Army, Headquarters. Safety - Ionizing Radiation Protection (Licensing, Control, Transportation, Disposal, and Radiation Safety). AR 385-11, Washington, D.C.

APPENDIX A

TRANSPORT GROUPS FOR SELECTED RADIONUCLIDES

APPENDIX A

TRANSPORT GROUPS FOR SELECTED RADIONUCLIDES

(49 CFR 173.390, 1980)

Element	Atomic No.	Symbol	Transport Group I	II	III	IV	V	VI	VII
Actinium	89	Ac	^{227}Ac ^{228}Ac						
Americium	95	Am	^{241}Am ^{243}Am						
Antimony	51	Sb			^{124}Sb ^{125}Sb	^{122}Sb			
Argon	18	Ar		^{41}Ar			^{41}Ar(a)	^{37}Ar	
Arsenic	33	As				^{73}As ^{74}As ^{76}As ^{77}As			
Astatine	85	At			^{211}At				
Barium	56	Ba		^{133}Ba	^{140}Ba	^{131}Ba			
Berkelium	97	Bk	^{249}Bk						
Beryllium	4	Be				^{7}Be			
Bismuth	83	Bi		^{210}Bi	^{207}Bi ^{212}Bi	^{206}Bi			
Bromine	35	Br				^{82}Br			
Cadmium	48	Cd			^{115m}Cd	^{109}Cd ^{115}Cd			
Calcium	20	Ca				^{45}Ca ^{47}Ca			
Californium	98	Cf	^{249}Cf ^{250}Cf ^{252}Cf(b)						

(a) Uncompressed (means at a pressure not exceeding 14.7 psi absolute).
(b) ^{252}Cf not more than 2 Ci in special-form type.

Element	Atomic No.	Symbol	Transport Group I	II	III	IV	V	VI	VII
Carbon	6	C				^{14}C			
Cerium	58	Ce				^{141}Ce ^{143}Ce			
					^{144}Ce				
Cesium	55	Cs				^{131}Cs			
					^{134m}Cs ^{134}Cs				
						^{135}Cs ^{136}Cs			
					^{137}Cs				
Chlorine	17	Cl			^{36}Cl				
						^{38}Cl			
Chromium	24	Cr				^{51}Cr			
Cobalt	27	Co			^{56}Co				
						^{57}Co ^{58m}Co ^{58}Co			
					^{60}Co				
Copper	29	Cu				^{64}Cu			
Curium	96	Cm	^{242}Cm ^{243}Cm ^{244}Cm ^{245}Cm ^{246}Cm						
Dysprosium	66	Dy			^{154}Dy				
						^{165}Dy ^{166}Dy			
Erbium	68	Er				^{169}Er ^{171}Er			
Europium	63	Eu			^{150}Eu	^{152m}Eu			
					^{152}Eu				
				^{154}Eu					
						^{155}Eu			
Fluorine	9	F				^{18}F			
Gadolinium	64	Gd				^{153}Gd ^{159}Gd			
Gallium	31	Ga			^{67}Ga				
						^{72}Ga			
Germanium	32	Ge				^{71}Ge			
Gold	79	Au			^{193}Au ^{194}Au ^{195}Au				
						^{196}Au ^{198}Au ^{199}Au			
Hafnium	72	Hf				^{181}Hf			

Element	Atomic No.	Symbol	Transport Group I	II	III	IV	V	VI	VII
Holmium	67	Ho				^{166}Ho			
Hydrogen	1	H(a)							
Indium	49	In			^{114m}In	^{113m}In ^{115m}In ^{115}In			
Iodine	53	I			^{124}I ^{125}I ^{126}I ^{129}I ^{131}I ^{133}I	^{132}I ^{134}I ^{135}I			
Iridium	77	Ir			^{192}Ir	^{190}Ir ^{194}Ir			
Iron	26	Fe				^{55}Fe ^{59}Fe			
Krypton	36	Kr			^{85m}Kr ^{85}Kr		^{85m}Kr(b)	^{85}Kr(a)	
				87Kr			^{87}Kr(b)		
Lanthanum	57	La				^{140}La			
Lead	82	Pb		^{210}Pb ^{212}Pb		^{203}Pb			
Lutetium	71	Lu			^{172}Lu	^{177}Lu			
Magnesium	12	Mg			^{28}Mg				
Manganese	25	Mn				^{52}Mn ^{54}Mn ^{56}Mn			
Mercury	80	Hg				^{197m}Hg ^{197}Hg ^{203}Hg			
Mixed Fission Products		MF-P		MF-P					
Molybdenum	42	Mo				^{99}Mo			
Neodymium	60	Nd				^{147}Nd ^{149}Nd			

(a) For ^{3}H, see tritium.
(b) Uncompressed (means at a pressure not exceeding 14.7 psi absolute).

Element	Atomic No.	Symbol	Transport Group I	II	III	IV	V	VI	VII
Neptunium	93	Np	^{237}Np ^{239}Np						
Nickel	28	Ni			^{56}Ni	^{59}Ni ^{63}Ni ^{65}Ni			
Niobium	41	Nb				^{93m}Nb ^{95}Nb ^{97}Nb			
Osmium	76	Os				^{185}Os ^{191m}Os ^{191}Os ^{193}Os			
Palladium	46	Pd				^{103}Pd ^{109}Pd			
Phosphorus	15	P				^{32}P			
Platinum	78	Pt				^{191}Pt ^{193}Pt ^{193m}Pt ^{197m}Pt ^{197}Pt			
Plutonium	94	Pu	^{238}Pu ^{239}Pu ^{240}Pu ^{241}Pu ^{242}Pu						
Polonium	84	Po	^{210}Po						
Potassium	19	K			^{43}K	^{42}K			
Praseodymium	59	Pr				^{142}Pr ^{143}Pr			
Promethium	61	Pm				^{147}Pm ^{149}Pm			
Protactinium	91	Pa	^{230}Pa ^{231}Pa	^{233}Pa					
Radium	88	Ra	^{226}Ra ^{228}Ra	^{223}Ra ^{224}Ra					
Radon	86	Rn		^{222}Rn		^{220}Rn			
Rhenium	75	Re				^{183}Re ^{186}Re ^{187}Re ^{188}Re Re(a)			

(a) Natural.

Element	Atomic No.	Symbol	Transport Group I	II	III	IV	V	VI	VII
Rhodium	45	Rh				^{103m}Rh ^{105}Rh			
Rubidium	37	Rb				^{86}Rb ^{87}Rb Rb(a)			
Ruthenium	44	Ru			^{106}Ru	^{97}Ru ^{103}Ru ^{105}Ru			
Samarium	62	Sm			^{145}Sm ^{147}Sm	^{151}Sm ^{153}Sm			
Scandium	21	Sc			^{46}Sc	^{47}Sc ^{48}Sc			
Selenium	34	Se				^{75}Se			
Silicon	14	Si				^{31}Si			
Silver	47	Ag			^{110m}Ag	^{105}Ag ^{111}Ag			
Sodium	11	Na			^{22}Na	^{24}Na			
Strontium	38	Sr		^{90}Sr	^{89}Sr ^{91}Sr	^{85m}Sr ^{85}Sr ^{92}Sr			
Sulfur	16	S				^{35}S			
Tantalum	73	Ta			^{182}Ta				
Technetium	43	Tc				^{96m}Tc ^{96}Tc ^{97m}Tc ^{97}Tc ^{99m}Tc ^{99}Tc			
Tellurium	52	Te			^{129m}Te ^{131m}Te	^{125m}Te ^{127m}Te ^{127}Te ^{129}Te ^{132}Te			
Terbium	65	Tb			^{160}Tb				

(a) Natural.

Element	Atomic No.	Symbol	Transport Group I	II	III	IV	V	VI	VII
Thallium	81	Tl				^{200}Tl ^{201}Tl ^{202}Tl			
					^{204}Tl				
Thorium	90	Th	^{228}Th ^{230}Th ^{231}Th	^{227}Th					
				^{234}Th	^{232}Th Th(a)				
Thulium	69	Tm			^{168}Tm ^{170}Tm	^{171}Tm			
Tin	50	Sn			^{117m}Sn ^{121}Sn	^{113}Sn ^{125}Sn			
Tritium	1	^{3}H				^{3}H			^{3}H(b)
Tungsten	74	W				^{181}W ^{185}W ^{187}W			
Uranium	92	U	^{232}U	^{230}U ^{233}U(c) ^{234}U ^{236}U	^{235}U(c) ^{238}U U(a) U(d) U Depleted				
Vanadium	23	V			^{49}V	^{48}V			
Xenon	54	Xe		^{135}Xe	^{125}Xe ^{131m}Xe ^{133}Xe	^{131m}Xe(a)	^{135}Xe(e)	^{133}Xe	
Ytterbium	70	Yb				^{175}Yb			

(a) Natural.
(b) As a gas, as luminous paint, or absorbed on solid material.
(c) Fissile radioactive material.
(d) Enriched, radioactive material.
(e) Uncompressed (means at a pressure not exceeding 14.7 psi absolute).

Element	Atomic No.	Symbol	Transport Group I	II	III	IV	V	VI	VII
Yttrium	39	Y			^{88}Y	^{90}Y			
					^{91m}Y				
					^{91}Y				
						^{92}Y			
						^{93}Y			
Zinc	30	Zn				^{65}Zn			
						^{69m}Zn			
						^{69}Zn			
Zirconium	40	Zr				^{93}Zr			
					^{95}Zr				
						^{97}Zr			

APPENDIX B

PACKING PROCEDURES FOR SELECTED QUANTITIES OF RADIOACTIVE MATERIALS

B.1 Procedures for Limited-Quantity Packages

B.2 Procedures for Low-Specific-Activity Packages

B.3 Procedures for Type A Quantity Packages

Note: To obtain instructions for the packaging of radioactive waste for disposal, contact HQ ARRCOM, ATTN: DRSAR-SF, Health Physicist, Rock Island, Illinois 61229. Telephone (309) 794-3383; FTS 367-3483; AUTOVON 793-4942.

APPENDIX B

PACKING PROCEDURES FOR SELECTED QUANTITIES OF RADIOACTIVE MATERIALS

Packaging procedures are described below for limited-quantity, low-specific-activity, and Type A quantity packages. For all three types, the following information on containers applies.

The primary container is the first container into which radioactive material is placed. The next or inner container should be strong, leaktight, and nonbreakable. For limited-quantity and LSA shipments (but not for shipments of Type A quantities), the primary container may be used as the inner container provided that it is strong, leaktight, and nonbreakable, and that the radioactive material is in solid form. For limited-quantity and LSA shipments, the outer container should be a fiberboard box, wooden box, metal can, or any container approved for Type A quantities of radioactive material (see Appendix C). For Type A shipments, only approved Type A containers may be used.

B.1 PROCEDURES FOR LIMITED-QUANTITY PACKAGES

Packaging of Solid Radioactive Material

1. Place the primary container in the inner container and add enough packing material to firmly secure the primary container in the inner container.
2. Mark the inner container with the words "Radioactive Material."
3. Place the inner container in the outer container and add enough packing material or bracing material to firmly secure the inner container and keep it centered in the outer container. (Free-flowing material such as vermiculite or absorbents is not suitable.)
4. Close the outer container and secure the closures to provide a tight package.
5. Apply appropriate address label.

Packaging of Liquid Radioactive Material

1. Place the primary container in a leaktight, strong, and corrosion-resistant inner container.
2. Add enough absorbent material to absorb at least twice the quantity of liquid radioactive material and to keep the primary container secure and centered in the inner container.
3. Mark the inner container with the words "Radioactive Material."
4. Place the inner container in the outer container and add enough packing or bracing material to firmly secure the inner container and keep it centered in the outer container.
5. Close the outer container and secure the closures to provide a tight package.
6. Apply appropriate address label.

Marking and Labeling

1. DOT-specification marking and labeling are not required for either off-site or onsite shipments.
2. The marking "Radioactive Material" on the inner conainer, as specified above, is required for both onsite and offsite shipments.
3. In addition, labeling specified in 49 CFR is required for shipments that have additional hazardous material mixed with or as part of the radio-active material.

B.2 PROCEDURES FOR LOW-SPECIFIC-ACTIVITY PACKAGES

Packaging of Solid Radioactive Material

1. Place the primary container in the inner container and add enough packing material to firmly secure the primary container in the inner container.
2. Place the inner container in the outer container and add enough packing material or bracing material to firmly secure the inner container and keep it centered in the outer container. (Free-flowing material such as vermiculite or absorbents is not suitable.)

3. Mark the inner and outer containers with the words "Radioactive - LSA."
4. Close the outer container and secure the closures to provide a tight package.
5. Apply appropriate address label.

Packaging of Liquid Radioactive Material

1. Place the primary container in a leaktight, strong, and corrosion-resistant inner container.
2. Add enough absorbent material to absorb at least twice the quantity of liquid radioactive material and to keep the primary container secure and centered in the inner container.
3. Place the inner container in the outer container and add enough packing or bracing material to firmly secure the inner container and keep it centered in the outer container.
4. Mark the inner and outer containers with the words "Radioactive - LSA."
5. Close the outer container and secure the closures to provide a tight package.
6. Apply appropriate address label.

Marking and Labeling

1. DOT specification marking and labeling are not required for either offsite or onsite shipments.
2. The markings specified above are required for both onsite and offsite shipments.
3. In addition, labeling specified in 49 CFR is required for shipments that have additional hazardous material mixed with or as part of the radioactive material.

B.3 PROCEDURES FOR TYPE A QUANTITY PACKAGES

Packaging of Solid Radioactive Material

1. Place the primary container in the inner container and add enough packing material to firmly secure the primary container in the inner container. Seal to make leaktight.
2. Place the inner container in the Type A outer container. Add enough bracing material and/or packing material to firmly secure the inner container and keep it centered. (Free-flowing material such as vermiculite or absorbents is not suitable.)
3. Close the outer container and secure closures to provide a tight package.

Packaging of Liquid Radioactive Material

1. Place the primary container in a leaktight, strong, and corrosion-resistant inner container.
2. Add enough absorbent material to absorb at least twice the quantity of liquid radioactive material and to keep the primary container secure and centered in the inner container; or, package the inner container inside a second leaktight, strong, corrosion-resistant inner container.
3. Place the inner container in the outer container and add enough packing or bracing material to firmly secure the inner container and keep it centered in the outer container.
4. Close the outer container and secure the closures to provide a tight package.

Marking Requirements

1. Onsite Shipments - Mark the container with the proper shipping name.
2. Offsite Shipments - Mark the Type A (outer) container with specification markings. These markings are also recommended for onsite shipments. For example:

   USA DOT 7A, Type A
   RADIOACTIVE MATERIAL
   Name of Shipper

Label Requirements

1. Onsite Shipments - Application of a radioactive material label to opposite sides of the package is recommended.
2. Offsite Shipments - Apply the appropriate radioactive white-I, yellow-II, or yellow-III label. List the radionuclide quantities in curies on all labels. Include the transport index on yellow-II and yellow-III labels.
3. Apply a tamper-proof seal to provide a means of determining whether unauthorized persons have tampered with the package.

APPENDIX C

EXAMPLES OF DOT-SPECIFICATION CONTAINERS

C.1 DOT Specification 17C Steel Drum (5-gallon)

C.2 DOT Specification 17H Steel Drum (30-gallon)

C.3 DOT Specification 12B-65 Fiberboard Box

APPENDIX C

EXAMPLES OF DOT-SPECIFICATION CONTAINERS

C.1 DOT SPECIFICATION 17C STEEL DRUM (5-gallon)

The Spec. 17C 5-gallon pail (see Figure 9.5) is authorized as an outer container for Type A quantities of solid radioactive material in normal or special form (49 CFR 178.115). Its dimensions are: 1) interior: 11-1/4-in. ID x 12-1/2-in. usable inside height; 2) exterior: 12-in. OD x 13-in. overall outside height. Specifications and restrictions for its use are as follows:

1. Authorized gross weight: 100 lb.
2. Any bulky equipment with sharp corners, protrusions, etc., must be securely positioned within drum.
3. Gasket material must have minimum operating range of -40°F to +130°F.

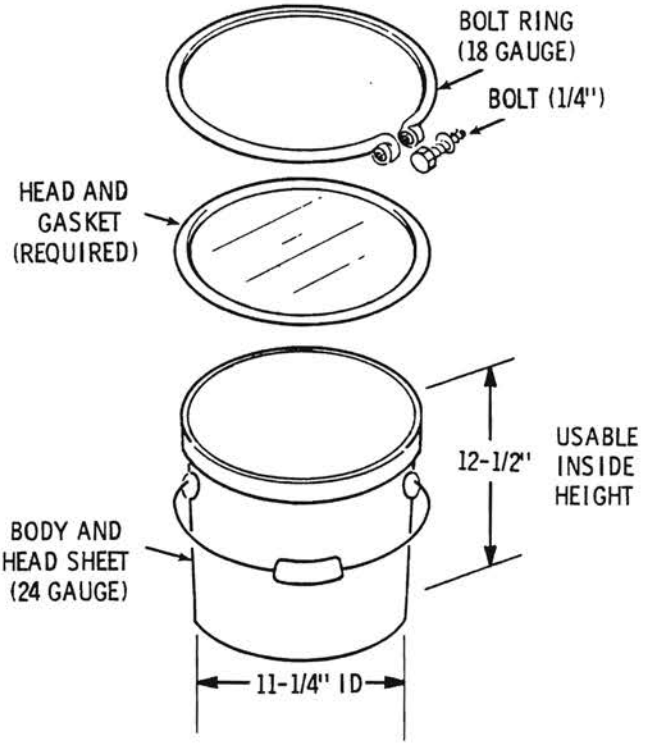


FIGURE 9.5. DOT Specification 17C Steel Drum (5-gallon)

C.2 DOT SPECIFICATION 17H STEEL DRUM (30-gallon)

The Spec. 17H 30-gallon steel drum (see Figure 9.6) is authorized as an outer container for Type A quantities of solid radioactive material in normal or special form (49 CFR 178.118). Its dimensions are: 1) interior: 18-1/4-in. ID x 28-in. usable inside height; 2) exterior: 20-in. OD x 29-1/2-in. overall outside height. Specifications and restrictions for its use are as follows:

1. Authorized gross weight: 500 lb.
2. Any bulky equipment with sharp corners, protrusions, etc., must be securely positioned within drum.
3. Gasket material must have minimum operating range of -40°F to +130°F. If sponge rubber gaskets are used, minimum of 1/2 in. required.

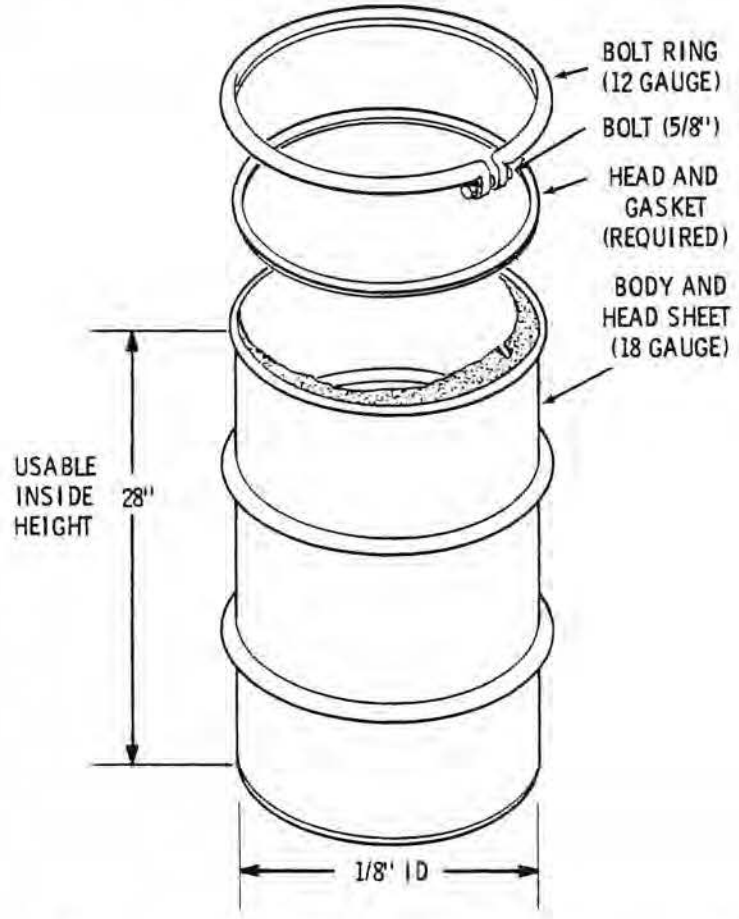


FIGURE 9.6. DOT Specification 17H Steel Drum (30-gallon)

C.3 DOT SPECIFICATION 12B-65 FIBERBOARD BOX

The Spec. 12B-65 fiberboard box (see Figure 9.7) is authorized as an outer container for Type A quantities of solid radioactive material in normal or special form (49 CFR 178.205). Radioactive material must further be contained within inner packaging(s) capable of withstanding the 4-ft drop test (49 CFR 173.398(b)(3)(iv)) unless suitable packaging materials are used to protect the inner contain. The box has an inner cushioning of rubberized horsehair in 1- and 2-in. layers. Two pieces of 3-in. plastic tape, run perpendicularly around the entire box, are used to close it.

The exterior dimensions of assembled and sealed boxes that have been tested and determined to meet Spec. 7A criteria are given below, along with the authorized gross weight of the contents for each box size.

Box Size (in.)			Authorized Gross Weight (lb)
Width	Height	Length	
12-1/2	12-1/2	9-1/2	26
12-1/2	12-1/2	13	26
14-1/2	17-1/2	28-1/2	65
18-1/2	13-1/2	18-1/2	41
18-1/2	18-1/2	19	65
24-1/2	24-1/2	24-1/2	65
24-1/2	12-1/2	24-1/2	65

Additional specifications and restrictions for the use of the box are as follows:

1. Any bulky equipment with sharp corners, protrusions, etc., must be securely positioned within the box.
2. All configurations are not authorized for air transport unless the inner container used is capable of withstanding the reduced-pressure test (49 CFR 173.398(b)(2)(iii)).
3. All configurations require an inner container capable of withstanding the 4-ft drop test (CFR 49 §173.398(b)(3)(ii)) or the use of suitable packaging materials to protect the inner container.

4. All configurations require an inner container capable of withstanding the penetration test (49 CFR 173.398 (b)(3)(iv)) or the use of suitable packaging materials to protect the inner container.

5. Stacking should be controlled/limited to the performance standard of five times the gross weight (5 x 65 lb = 325 lb) unless consideration is given to the effect on the inner container.

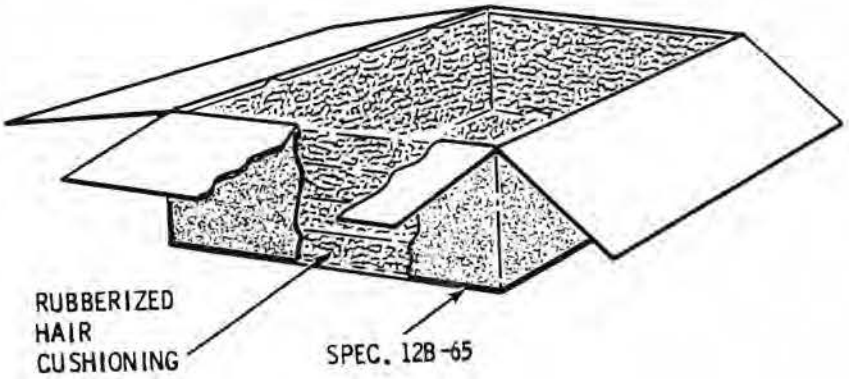


FIGURE 9.7. DOT Specification 12B-65 Fiberboard Box

CHAPTER 10. MANAGEMENT OF LOW-LEVEL RADIOACTIVE WASTE

CHAPTER 10. MANAGEMENT OF LOW-LEVEL RADIOACTIVE WASTE

Low-level radioactive waste is waste that contains 1) low enough levels of beta-gamma activity so that no special provisions must be made for heat removal, and 2) low enough levels of penetrating radiation so that minimal or no biological shielding or remote handling is necessary for personnel protection. Low-level waste is generally considered to contain less than 100 nCi of transuranic alpha emitters (uranium, thorium, etc.) per gram of waste. The handling, storage, and disposal of low-level radioactive waste must conform to strict requirements imposed by the Department of the Army (DA), the Nuclear Regulatory Commission (NRC), the Department of Transportation (DOT), and the operators of waste burial grounds.

This chapter provides guidance for persons who generate low-level radioactive waste and for those responsible for its handling, storage, and disposal. Topics covered include the generation and collection of waste, facilities for storing it, and procedures for reducing waste volumes and obtaining DA assistance in waste disposal. Further questions about the management of low-level waste should be directed to HQ, ARRCOM, ATTN: DRSAR-SF, Health Physicist, Rock Island, Illinois 61299. Telephone calls can also be placed to (309) 794-3383; FTS 367-3483; or AUTOVON 793-4942.

Section 10.1 MINIMIZING THE GENERATION OF WASTE

The use of radioactive material should be planned so that a minimum amount of radioactive waste is generated. For example, when a procedure requires the use of radioactive material, a dry run using nonradioactive material can eliminate errors that might cause contamination and create considerable waste. The smallest quantity of radioactive material needed to effectively perform a task should always be used.

The volume of radioactive waste can be reduced if nonradioactive and radioactive wastes are separated and not discarded together. Solid dry wastes

that may be contaminated should be surveyed and the nonradioactive portions discarded by conventional methods. When a device is being discarded, any radioactive components should be removed if the device can be disassembled safely and the disassembly is authorized by the NRC license or DA permit. Every effort should be made to decontaminate contaminated property before it is disposed of. However, the volume of waste that would be generated by the decontamination procedure should be considered before low-cost items are decontaminated (see Chapter 7).

Section 10.2 COLLECTION OF WASTE

The total quantity of radioactive material disposed of into sanitary sewage systems, the air, or nearby streams as a result of all activities at an installation must not exceed the quantities for a single licensee given in 10 CFR 20, or the quantity limitations established by applicable regulatory agencies. Individual users of radioactive material must not dispose of waste directly by these methods unless specifically authorized by the Radiation Protection Officer (RPO). Instead, each user should collect any low-level wastes according to the guidelines in this section.

When wastes are being collected at a facility, the radioactive waste should be separated from the nonradioactive waste. Wastes that are taken from a radiation area should be presumed to be radioactive unless shown to be otherwise. This is particularly true in hot laboratories, where paper tissue and even writing paper may become significantly contaminated. Radioactive wastes should be segregated into classes of material so that all constituents of any one batch can be dealt with in the same way. They should be collected in suitable containers for processing and disposal by the RPO or a designated representative.

10.2.1 Segregation of Radioactive Waste

Characterization of low-level radioactive waste is important for proper waste handling and processing for final disposal. Characterization includes identification of the physical form of the waste, the type and half-life of

radionuclides present, the total activity and/or specific activity, and other properties of the waste such as its volatility, explosiveness, and toxicity.

Once characterized, wastes should be collected according to type by each user of radioactive materials. Procedures for segregating and collecting wastes should be developed by the RPO and provided to all individuals who may generate radioactive waste as a result of their work. The procedures should cover the segregation of wastes by half-life and by the characteristics described below. The waste collected under each category can be further separated by whether it is combustible or compactible.

A. Half-Life. Waste containing short-lived radionuclides (those with a half-life ($t_{1/2}$) shorter than 30 days) should be collected separately from waste containing long-lived radionuclides (those with a half-life longer than 30 days). Short-lived material can usually be stored away from work areas for 10 half-lives of the longest-lived radionuclide in the material and then discarded as nonradioactive material. It must be surveyed before disposal by conventional methods. Long-lived material should be processed for disposal as radioactive waste.

B. Biological Waste. Biological waste, which originates primarily from medical and research facilities, normally undergoes decomposition by microorganisms, producing foul-smelling matter. Such material requires freezer storage.

(1) Solid. Solid biological waste includes radioactively contaminated animal carcasses, fecal matter, soiled animal bedding, and plant by-products. Personnel working with animals should be aware of radiation levels and of the excretion routes for various radiochemicals and drugs (National Council on Radiation Protection and Measurements (NCRP) Report No. 48, 1976). Animals that are used in studies of radioactive materials should not be petted or groomed, and their carcasses should not be hand-carried if a radiation overexposure to the hands or body of the person carrying them may result. Remote handling and storage is advised (TM 3-261).

(2) Liquid. Liquid biological waste includes radioactively contaminated blood, urine, and culture media. Because biological waste should be stored frozen, containers should be capable of withstanding temperature extremes

without breaking and should be filled about three-quarters full to allow for expansion of the contents. Polyethylene containers are preferred. Personnel dealing with liquid biological wastes should consider not only the radiological hazards and the need to provide radiation protection, but also the potential chemical and biological hazards that may be associated with the wastes.

C. Nonbiological Waste. Nonbiological waste is any radioactively contaminated waste that, under ordinary circumstances, does not undergo decomposition by microorganisms.

(1) Solid. Solid nonbiological waste makes up the major portion of low-level radioactive waste. It includes radioactively contaminated glassware, protective clothing, gloves, paper, metal scraps, syringes, filters, sealed sources, and equipment or equipment components (compasses, meters, electron tubes, etc.). Depleted uranium, either as an ore or in metal form, also falls into this category. If a device with a solid source is not internally or externally contaminated, it should be handled in a manner that prevents its contamination. For example, it should not be placed in the same collection container as a pair of contaminated gloves.

(2) Liquid. Not all liquids are disposed of in the same way; therefore, liquid nonbiological waste should be segregated into aqueous and nonaqueous waste. Aqueous waste--any waste in which water is the primary solvent--includes water used to decontaminate material or personnel, and solutions of radioactive material used in a laboratory. Nonaqueous waste is any liquid in which water is not the primary solvent.

Any chemically reactive liquids should be further segregated and identified. Organic liquids (those containing carbon compounds) should be segregated from aqueous solutions to prevent the possibility of violent reactions. Nitric acid and alcohol, for example, if disposed of in the same vessel, could react together and cause an extensive spread of contamination. Unless special arrangements are made with the RPO, individuals who generate strongly acidic or basic waste solutions should neutralize or dilute them enough so that they will not cause violent chemical reactions or release strong fumes and vapors. In the case of organic solvents, especially those that are highly volatile, appropriate precautions should be noted on the waste container.

When the methods used to dispose of liquid wastes include absorption of the liquid or ion exchange processes, the potential for chemical interactions that could affect the process should be evaluated. Types of liquid wastes that could cause adverse effects on the processing of the waste include acidic or basic solutions; liquids containing complexing or wetting agents; and liquids containing certain detergents. Precautions must be taken to prevent the accidental processing of incompatible liquid wastes.

D. Scintillation Vials. Small glass or plastic vials containing scintillation fluids and low levels of radioactively labeled compounds may be handled as an entity; the contents of the vials need not be transferred to a waste container. The vials should be packaged (preferably in their original cartons) to avoid breakage, and the box should be properly labeled.

10.2.2 Containers for Collection and Temporary Storage of Waste

Containers used for the collection and temporary storage of radioactive waste should be made of materials that will not rust or corrode from contact with the wastes stored in them. The lids of the containers must be easy to open so that the containers do not tip over when the lids are being removed.

Each container of radioactive waste should be painted bright yellow and marked "Caution - Radioactive Material." It should be labeled with enough information to permit accurate identification of the waste it contains. This information, which should be noted on the label at the time the waste is placed in the container, should include:

1. the name of the waste generator
2. the date
4. the pH of a waste solution
3. the chemical name of the waste material
5. the isotope(s) contained in the waste
6. the activity level
7. any information on the biological or chemical hazards associated with the waste.

Waste containers should be checked periodically to ensure that radiation levels are not excessive, that outside surfaces are free of contamination, and

that corrosion or rust is not weakening the container. If a waste container that is being used for the collection of wastes develops a high external-radiation exposure level or becomes externally contaminated, it should not be kept even temporarily at the user's location, but should be moved immediately to the storage site for radioactive wastes. A container that is corroding and losing its integrity should be placed inside a second container before being moved.

Individuals who generate waste should notify the RPO whenever a container is filled and ready for removal. The RPO should remove the waste and place it in a centralized area for temporary storage and consolidation. Containers should not be moved unless they are labeled and the waste is contained in accordance with installation requirements.

A. Containers for Biological Waste. Solid biological waste must be sealed in plastic bags and frozen. Liquid biological waste should be stored in plastic containers that can be frozen without breaking. Glass containers are not acceptable (TM3-261). Biological wastes are packed in lime for shipping.

B. Containers for Nonbiological Solid Waste. Solid waste must be sealed in plastic bags. It can be stored in a metal waste can with a plastic liner and a lid that operates by a step-pedal. When the waste is to be moved, it must be packaged so that pipettes, hypodermic needles, and other sharp objects cannot penetrate through the plastic bag.

C. Containers for Nonbiological Liquid Waste. Glass containers should not be used to store liquid waste. Aqueous waste may be kept in polypropylene carboys or jugs. Nonaqueous waste (organic solvents, acids, and bases) may be kept in metal solvent cans or in plastic containers if the liquid will not dissolve the plastic.

Section 10.3 FACILITIES FOR THE STORAGE OF WASTE

A facility should be designated for the centralized storage of radioactive wastes until they are shipped for processing or burial.

10.3.1 Site Selection

When a storage facility is being selected, whether an existing structure or a new structure, the RPO or the individual responsible should ensure that the following guidelines, in addition to those in Chapter 8, are met:

1. The facility is close to the point of origin of the waste but away from main areas of personnel traffic or areas where routine access is required.
2. The facility is weatherproof and has adequate ventilation.
3. Enough storage space is provided to allow for variations in shipping schedules and, if possible, to store short-lived materials (those with a half-life shorter than 30 days) while they decay.
4. Separate storage compartments are provided for combustible liquids (for fire prevention).
5. Means of handling wastes efficiently are provided, to minimize personnel exposures.
6. The radiation dose limits for the unrestricted area around the facility will not be exceeded.

10.3.2 Control Procedures

To keep personnel exposures to a minimum and to protect the general public, only individuals responsible for storing or shipping waste should have access to the waste storage facility. The wastes should be kept segregated by type, with higher-level waste placed far from the facility entrance to reduce the exposure to personnel who enter the area. As waste is brought into or taken out of storage, the amount and type of the waste moved, the date, and the name of the user or shipper should be entered in a log book. Personnel monitoring should be provided to ensure contamination control.

Section 10.4 VOLUME REDUCTION

Reducing the volume of low-level waste has the following benefits:

1. It increases the stability of the waste form.
2. It minimizes the possibility that radionuclides will be released to the environment during interim storage, transportation, and burial.
3. It leads to savings in transportation and burial costs, which are dependent on waste volume.
4. It reduces the exposure of personnel handling the waste.

As discussed earlier, volume reduction should be accomplished primarily by each person minimizing the amount of waste generated. Wastes that have been generated can be reduced in volume by solidification, compaction, and incineration.

Volume reduction processes can be carried out most economically at central waste-consolidation facilities to which many installations or sites ship their radioactive wastes for treatment before final disposal. The use of volume reduction equipment at an Army installation requires an NRC license and a DA authorization or permit.

10.4.1 Solidification

Many burial sites require that the wastes they handle meet certain physical forms. Low-level liquid wastes must be converted to a solid that will not leach. Loose, dry residues from incinerators or dryers must be bound together into a solid waste form.

A variety of methods are used to solidify wastes and reduce their volume. Aqueous solutions are treated by crystallization and dehydration. Crystallization is the removal of water, usually by evaporation, which results in a slurry of precipitated solids mixed with a saturated solution. The slurry is then mixed with a setting agent such as cement. Dehydration is the removal of all the water from liquid wastes, leaving a residue of solids. Aqueous liquids and dry residues from incinerators and dryers can be mixed with a binding agent to form a solid waste. Conventional setting and binding agents are cement, bitumen, glass, and urea-formaldehyde; experimental materials include vinyl esters, polyethylene, epoxy resins, and an inorganic binder.

10.4.2 Compaction

Compaction (the removal of excess air) is the most widely used method of volume reduction for dry, nonbiological wastes that are not combustible. Compaction includes compressing the waste into a final disposal container (such as a 208-liter drum) and baling the compressed waste with bands before packaging it. Items that are currently compacted in the commercial fuel cycle include high-efficiency particulate air (HEPA) filters and old contaminated drums.

Before compaction, the waste should go through some pretreatment. Compactible and noncompactible items should be separated. Hazardous materials (such as explosives) and materials containing free liquid should be removed. Items that would otherwise be too large for the compactor can be shredded using knife cutters or hammermills. For example, equipment and metal can be packaged as is or shredded in a hammermill and compacted.

A typical compactor for low-level waste consists of a hydraulic system with a vertical ram, a contoured support plate, a frame, a safety enclosure, and automatic controls. These drum compactors should be located in protective enclosures, which prevent the escape of airborne particulate matter. A hood or shroud around the drum opening, with a HEPA filter and an exhaust blower, serves to control particulates. Some drum compactors incorporate a metal inner sleeve to protect the drum walls from the pressure of the ram and from rigid metal objects.

10.4.3 Incineration

Incineration is the removal of combustible material in radioactive waste. Water and air are removed at the same time. The types of incinerators available for radioactive-waste processing include controlled-air incinerators, fluidized-bed incinerators, and rotary kilns. These systems differ in operating temperatures, waste residence times, chamber turbulence, and amount of oxygen used. Each incineration system requires specific methods of waste pretreatment, feeding, ash removal, and off-gas treatment.

All incinerators for radioactive waste must have an off-gas system to keep particulate and gaseous effluents within NRC, Environmental Protection

Agency (EPA), and state limits. These off-gas systems result in an additional radioactive waste stream that must be considered.

The main advantage of incineration as a volume reduction process is the uniform end product, ash, which is easy to solidify and thus minimizes the problems associated with the disposal of a wide range of materials. The main disadvantage is the high initial cost; because a relatively large volume of waste material must be generated to make the procedure cost-effective, incineration is not economical for most sites.

Section 10.5 AUTHORIZED DISPOSAL PROCEDURES

The DA program for the disposal of low-level radioactive waste is managed by HQ, ARRCOM, Rock Island, Illinois. The authority for world-wide management of the program is assigned in AR 385-11.

Low-level waste that cannot be disposed of locally because of local restrictions is disposed of by land burial in Barnwell, South Carolina, or Richland, Washington, by commercial radioactive-waste-disposal firms under contract with HQ, ARRCOM. Under certain conditions, waste shipments are sent to a collecting point operated by a waste disposal broker or the Army. At the collecting point, they are consolidated and ultimately disposed of.

10.5.1 Requests for Disposal Instructions

The RPO is responsible for requesting disposal instructions from the Commander, US Army Armament Materiel Readiness Command, ATTN: DRSAR-DSM-D, Rock Island, Illinois 61299. The request can be made by letter or message.

Requests for disposal instructions must contain the following information:

1. nomenclature, national stock number, and serial numbers
2. physical descriptions of the items to be disposed of, including:
   a. whether solid, liquid, or gas
   b. the quantity per stock number and, if gas, the volume under standard pressure and temperature

c. the shipping weight (pounds) and volume (measured to the nearest cubic foot)
d. the number of shipping containers
e. the shipping permit or waiver number
f. the transport group
g. the package specification
h. the labels used

3. chemical and radioisotope description, including:
   a. the hazardous chemicals present
   b. for liquids, the solvent present
   c. the radioisotopes present

4. radioactivity and radiation measurement, including:
   a. the millicuries of activity of each radioisotope; for special nuclear material, give the number of grams; for source material, list the quantity in pounds
   b. maximum radiation dose rates (mrem/hr) at the surface and 1 meter from the surface of the package
   c. the classification, basis for classification, and procedures for declassification
   d. special instructions or requests for unique service, such as return of the containers
   e. the name and telephone number of the responsible person to contact for additional information
   f. remarks, if appropriate.

Requests for technical information or assistance should be submitted to the Commander, ARRCOM, ATIN: DRSAR-SF, Rock Island, Illinois 61299. Telephone requests can be made by calling (309) 794-3383/4728; FTS 367-3383/4728; or AUTOVON 793-3383/4728.

10.5.2 Shipping Instructions

Shipping instructions will be furnished by HQ, ARRCOM, in reply to requests for disposal instructions. Each request will be handled as a separate action, and the instructions will include the following:

1. an ARRCOM-assigned control number, which will serve as the identification for each request
2. the address of the shipping destination as determined by HQ, ARRCOM; the destination may be a land burial site or a collection/consolidation point
3. specific marking, packaging, and transportation instructions.

Because safety concerns and burial criteria change periodically, special instructions will also be furnished.

10.5.3 Onsite Assistance

Radioactive-waste shipments may be audited by HQ, ARRCOM, at the shipper's installation prior to shipment. Some audits require that an ARRCOM audit team be onsite to supervise the packaging and loading of the radioactive material. Requests for onsite assistance should be addressed to Commander, US Army Armament Materiel Readiness Command, ATTN: DRSAR-SF, Rock Island, Illinois 61299.

REFERENCES

National Council on Radiation Protection and Measurements (NCRP). 1976. Radiation Protection for Medical and Allied Health Personnel. NCRP 48, Washington, D.C.

U.S. Code of Federal Regulations. 1982. Title 10, Part 20, "Standards for Protection Against Radiation." U.S. Government Printing Office, Washington, D.C.

U.S. Department of the Army, Headquarters. Handling and Disposal of Unwanted Radioactive Material. TM 3-261, Washington, D.C.

U.S. Department of the Army, Headquarters. Safety - Ionizing Radiation Protection (Licensing, Control, Transportation, Disposal, and Radiation Safety). AR 385-11, Washington, D.C.

CHAPTER 11. RADIATION ACCIDENTS AND EMERGENCY PREPAREDNESS

TABLES

CHAPTER 11. RADIATION ACCIDENTS AND EMERGENCY PREPAREDNESS

A radiological emergency is any sudden or unforeseen situation in which damage to persons or property, or interruption in operations, has occurred or is imminent unless corrective actions are taken. The severity of both an accident and its effects can be decreased if procedures are followed, engineered controls are used, and corrective and protective actions are taken.

Planning for radiological emergencies can uncover problems that, if corrected, will decrease the likelihood of an accident. Therefore, a plan for responding to abnormal occurrences should be developed and maintained for each individual operation involving radioactive materials. Each plan will vary from others accordings to the specifics of the operation. The magnitude of the emergency planning needed at an installation and the notification, reporting, and investigative procedures required in the event of an accident depend on the potential hazards at each facility and the types of accidents that may occur.

In this chapter, radiological accidents are identified and classified, guidance is provided on how to prepare for potential accidents by developing an emergency preparedness plan and how to maintain a state of emergency preparedness, and accident reporting and investigative procedures are reviewed. Emergency preparedness is a full-time specialty of health physics that requires training and experience. This chapter is intended to introduce the Radiation Protection Officer (RPO) to emergency preparedness. The assistance of a trained specialist should be sought for developig extensive plans and emergency responses.

Section 11.1 THE EMERGENCY PLAN

An emergency plan is a document that details the best response to an emergency situation, with primary cncern for protecting the health and safety of Army and civilian personnel and the general public. A comprehensive plan should contain the following key elements:

1. designation of responsibility for emergency planning
2. assessment of potential accidents
3. system for classifying emergencies
4. description of the emergency response organization
5. characterization of the installation and its facilities
6. description of activities authorized by the Department of the Army (DA) and the Nuclear Regulatory Commission (NRC)
7. procedures for implementing the emergency plan
8. response actions
9. description of facilities and equipment
10. description of offsite agreements and support capabilities
11. re-entry and recovery conditions.

The use of a checklist such as that presented in Appendix A can help ensure that all aspects of an emergency plan have been considered.

11.1.1 Responsibility for Emergency Planning

The commander of each installation is responsible for planning for and providing training for credible emergencies (AR 385-11). This duty may be delegated to an organization within the command that has the operational experience and technical abilities necessary to direct planning efforts. Personnel involved in emergency planning must have the authority to gather site-specific information, write procedures, and enter into discussions with offsite agencies. In many cases, the RPO and the Ionizing Radiation Control Committee (IRCC) are the logical choices for this duty. If the duty is delegated elsewhere, the RPO and the IRCC should be involved in at least the radiological assessment, control, and recovery aspects of emergency planning.

11.1.2 System for Classifying Emergencies

Emergency plans and procedures should be developed for all facilities where radioactive materials are handled, used, stored, or transported, regardless of quantity. However, formal documented emergency plans must be

prepared (DARCOM Disaster Control Plans (DCP), Annex E) if the quantities of radioactive materials exceed:

1. 1 μCi of radium
2. the quantities listed in Schedule B of 10 CFR 30.71
3. 6.8 kg of source material
4. 5 μg of special nuclear material.

Schedule B of 10 CFR 30.71 sets limits for byproduct materials. A portion of Schedule B (the more common byproduct materials) has been reproduced in Table 11.1.

TABLE 11.1. Significant Quantities of Byproduct Materials[a]

Byproduct Material	Microcuries	Byproduct Material	Microcuries
^{3}H (tritium)	1,000	^{115}Cd	100
^{14}C	100	^{115m}Cd	10
^{18}F	1,000	^{124}Sb	10
^{32}P	10	^{125}I	1
^{35}S	100	^{131}I	1
^{36}Cl	10	^{133}Ba	10
^{42}K	10	^{133}Xe	100
^{54}Mn	10	^{135}Xe	100
^{59}Fe	10	^{137}Cs	10
^{60}Co	1	^{144}Ce	1
^{65}Zn	10	^{147}Pm	10
^{85}K	100	^{148}Pm	10
^{90}Sr	0.1	^{197}Hg	100
^{90}Y	10	^{197m}Hg	100
^{99}Mo	100	^{198}Au	100
^{99}Tc	10	^{204}Tl	10
^{99m}Tc	100	^{210}Bi	1
^{109}Cd	10	^{210}Po	0.1
^{115}In	10		

(a) Excerpted from 10 CFR 30.71, Schedule B.

The scope of the emergency plan depends on the potential hazards of the maximum credible accident and other postulated accidents. The maximum credible accident is the accident that would cause the highest radiation exposures to onsite personnel and/or the public. Although the maximum credible accident poses the greatest threat, all potential accidents should be considered in the development of emergency plans. The presence of small quantities of radioactive materials may require only a few procedures and telephone numbers, with minimal supplies and equipment (e.g., ropes, signs, and survey meters). The presence of large quantities may require an extensive plan, many procedures, and facilities and equipment dedicated to an emergency response.

Assistance should be obtained for emergency planning, particularly if the installation does not have the resources to handle the identified credible accidents. Assistance may be available from Army emergency response teams or health physics specialists in emergency preparedness. If local personnel cannot identify such assistance, contact DARCOM or the office of the Surgeon General of the United States.

Four classes of emergency conditions that are frequently used in the nuclear industry to classify potential hazards--unusual event, alert, site emergency, and general emergency--are described in Table 11.2 (pages 11.10-11.11), based on NRC's NUREG 0654 (1980). The classes are defined in terms of onsite and offsite consequences and projected dose commitments and exposure rates at the boundary of the event site, which may be the door of a laboratory or a building, or a restricted-access gate on base. Army operations would typically encompass only the first two emergency classes: unusual event and alert. If a site emergency or general emergency that might cause the release of radioactive materials to offsite locations could occur at an installation, assistance should be sought in designing and developing emergency plans and procedures.

The following topics should be considered in the development of emergency plans:

1. the kinds of radioactive materials potentially released (so that responsive monitoring instrumentation can be identified)

2. the most important exposure pathways for these types of materials (so that the effect on the local population can be determined)
3. a definition of the area for which planning should be carried out (called the emergency planning zone (EPZ))
4. the potential duration of a release and the time available before exposures offsite are significant (so that protective actions can be decided upon).

Specific conditions, both actual and imminent, that require an emergency response are called emergency action levels (EALs) and are the basis for declaring an unusual event, an alert, or a higher classification of accident. When the EALs have been identified and documented, the procedures, facilities, and equipment required for a response can also be identified. Thus, the EALs can provide a framework for developing emergency procedures.

Another useful classification system (Brodsky 1980) groups commonly used radionuclides into eight groups based on the relative magnitudes of their maximum radiotoxicities. This system was presented in Chapter 1 of this manual. It can be useful in determining EALs and specifying subsequent actions.

11.1.3 Emergency Response Organization

The coordinated efforts of several organizations may be required to produce an adequate emergency response. In the emergency plan, one individual must be designated as having overall responsibility and authority for implementing and directing emergency procedures. Each support organization and its responsibilities must be identified, and persons responsible for each group must be identified by title or position, along with any alternates, to assure a 24-hr/day response. All individuals assigned responsibilities must have knowledge of and experience in radiological emergency preparedness.

Table 11.3 is a listing of the organizational support personnel who must be available at any installation and included in any plan, with brief example descriptions of their responsibilities. Site requirements may call for a more complex list or may allow two or more organizational functions to be consolidated. Several duties of key response personnel cannot be delegated. For instance, the emergency director cannot assign subordinates the responsibility

TABLE 11.2. Emergency Condition Classification Scheme (NUREG 0654)

Emergency Class	Description	Projected Dose Commitments and Exposure Rates at Event Site Boundary
Unusual Event	An event that is confined to facilities and personnel onsite but indicates the loss of one level of safety control or control boundary. Emergency response actions are limited to onsite areas. Examples are: laboratory spills, radiation overexposures, and contaminated personnel, rooms, or buildings. The purpose of the unusual event level is to alert plant personnel to unusual radiological conditions, to make management aware that administrative controls have been violated, and to initiate management actions to prevent recurrence.	Whole-Body (WB) dose commitment: <1 mrem Organ dose commitment: <3 mrem WB exposure rate: <0.5 mR/hr
Alert	An event that involves an actual or potential substantial reduction of the level of safety of personnel and the facility. Limited releases of hazardous materials to offsite locations may occur but are not expected to exceed applicable permissible limits. The purpose of an alert level is to ensure that 1) onsite (and possibly offsite) emergency response personnel are properly advised and available for activation if the situation becomes more serious, 2) that confirmatory radiation monitoring is performed, and 3) that response organizations are notified of emergency conditions.	WB dose commitment: 1 to 50 mrem Organ dose commitment: 3 to 150 mrem WB exposure rate: 0.5 to 50 mR/hr

TABLE 11.2. (continued)

Emergency Class	Description	Projected Dose Commitments and Exposure Rates at Event Site Boundary
Site Emergency	An event that involves actual or likely major failure of facility functions that are needed for the protection of onsite personnel, the public health and safety, and the environment. Releases of hazardous materials to offsite locations are likely or are occurring and have the potential to exceed applicable permissible limits. The site emergency may be declared if an emergency has occurred or is imminent. The purpose of the site emergency level is to ensure that 1) emergency control centers are manned, 2) appropriate monitoring teams are dispatched, 3) personnel required for determining onsite protective measures are at duty stations, 4) predetermined protective measures for onsite personnel are initiated, and 5) offsite officials and organizations are informed of the situation.	WB dose commitment: 50 mrem to 1 rem Organ dose commitment: 150 mrem to 3 rem WB exposure rate: 50 to 500 mR/hr
General Emergency	An event that involves actual or imminent substantial reduction of facility safety. Releases of hazardous materials to offsite locations are occurring or are expected to occur and exceed applicable permissible limits. The purpose of the general emergency level is 1) to initiate predetermined protective actions for onsite personnel, the public health and safety, and the environment, and 2) to provide continuous assessment of emergency conditions and exchange of information both onsite and offsite. Declaration of a general emergency will initiate major activation of DA-wide resources required to effectively mitigate the consequences of emergency conditions and ensure the protection of onsite personnel, the public health and safety, and the environment to the greatest extent possible.	WB dose commitment: >1 rem Organ dose commitment: >3 rem WB exposure rate: >500 mR/hr

TABLE 11.3. Minimum Organizational Support for Emergency Preparedness

Emergency Function	Normal Duty	Emergency Responsibility
Emergency Director (ED)	Commander of installation or Officer of the Day	Makes all decisions on how to respond to an accident based on recommendations from emergency support organization and guidance from the emergency plan. Authorizes nonprocedural actions for assessment, protective actions, and recovery. Declares appropriate emergency class.
Assistant Emergency Director	Anyone familiar with plan and designated by ED	Gathers information from various organizations and feeds pertinent information to ED.
Radiological Assessment and Control (RAC)	Radiation Protection Officer	Supervises measurement, assessment, and control of radiological conditions. Performs onsite and offsite dose calculations. Makes protective action recommendations to ED as needed.
Facilities Director	Same as routine operation	Consulted on building designs, utilities, equipment, operations, etc.
Offsite Activity Coordinator	Designated by ED; familiar with offsite agency agreements and capabilities	Keeps offsite agencies informed about situation and coordinates their actions with onsite response.
Public Information	Public relations spokesman	Releases emergency information to public as authorized by ED.
Security	Same as normal operations	Under command of ED, isolates affected areas, controls traffic and crowds. By agreement, assists in offsite control of traffic, crowds, access, etc.
Fire Safety, Search and Rescue	Same as normal operations	Assesses and controls nonradiological hazards; works with RAC on search and rescue; provides first aid medical treatment and fire control.

for declaring emergency classifications or recommending protective actions. Responsibilities that cannot be delegated must be identified in the plan.

11.1.4 Characterization of Installation and Facilities

The emergency plan should include a description of the principal characteristics of the installation. Approximate populations of onsite and offsite structures should be identified. Aerial photographs or site maps should be used to identify the location of facilities or areas relevant to emergency planning. These could include: 1) the location of population centers (office buildings, schools, barracks, stadiums, personnel housing); 2) the location of facilities that could present potential evacuation problems (hospitals, schools); 3) identification of primary routes for bringing in emergency equipment or for evacuating personnel or the public; 4) location of emergency support facilities (fire stations, hospitals with capability for handling patients with radioactive contamination); and 5) other sites of potential emergency significance (hazardous chemicals, gas lines).

Facilities in which radiological activities are conducted should be concisely described. The description should include confinement structures for handling and storing radioactive and other hazardous materials; auxiliary systems such as ventilation; radioactive waste management; and detection and alarm systems.

11.1.5 DA-Authorized and NRC-Licensed Activities

Work that involves radioactive materials and that is authorized by the Army and licensed by NRC should be described in the emergency plan. Included should be the location of the work; the type, form, and quantity of the radioactive materials used; the type of waste produced; and the individuals responsible for the activities.

11.1.6 Emergency Plan Implementation

The emergency plan should include detailed instructions for carrying out emergency response actions and information on required notifications.

A. Procedures. The detailed response procedures should include the following:

1. specific EALs and the actions planned in response to them
2. a statement of the responsibilities assigned to each individual, and which responsibilities may not be delegated
3. references to support documents and procedures that supplement the emergency plan.

The procedures should be developed to ensure that all positions will be manned and all appropriate emergency organizations will be operational in the event of an emergency.

B. Notification. When an emergency class is declared, prompt notification of personnel is vital to response. Methods and procedures for 24-hr/day notification of each organization that has an emergency response assignment are necessary. A site-wide notification system (i.e., public address or pageboy system) is useful in alerting site personnel; however, someone must confirm that response groups have been notified. A call list of key emergency response personnel and their alternates, and of DA and NRC contacts, should be part of the emergency plan, and one person or group should be designated to contact them at the direction of the emergency director. Contacts should be completed within 15 minutes of the declaration of an emergency class.

The methods of communication that will be used to notify onsite and offsite personnel must also be specified in the emergency plan, including a description of all primary and back-up notification equipment. Messages and announcements that are planned and written out in advance are useful and should be incorporated into the procedures to avoid delays and misunderstandings.

11.1.7 Response Actions

Emergency response actions fall into three general categories: assessment actions, corrective actions, and protective actions. Individuals who have emergency response assignments should be experienced in their assigned responsibilities and should have access to procedures that stipulate what actions should be taken. Procedures should be well written, easy to understand, and presented in a "cookbook" format, with space allotted for notes. Appendix A contains a sample checklist of procedures to be followed in the

event of a minor spill. Appendix B provides specific response actions and considerations for accidents involving exposure to individuals and for transportation accidents.

The following sections provide a synopsis of how the three categories of response actions should be treated in the emergency plan.

A. Assessment Actions. Responding to accident situations requires knowing both present and impending radiological conditions, which can be calculated using available information and supplemented with data obtained from radiological surveys. If insufficient information is available for making calculations, survey data alone may be used to determine emergency response actions.

For radiological surveys, instruments and equipment capable of measuring all anticipated conditions must be available and operational. The type and number of instruments needed depend on how extensive the onsite and offsite measurements will be. A program may be greatly simplified if only onsite response is required. An offsite capability requires thorough planning over a large area, special radiological equipment, and vehicles for transporting personnel and equipment.

Instruments must be capable of measuring the full range of anticipated radiation intensities and types. The specifications provided by vendors should be tested, as should each instrument's response to the 50-year environmental extremes recorded in each location.

Onsite parameters that must be measured are dose rate, contamination count rate, and the concentrations of radionuclides in air and effluents. Offsite parameters are the same except that meteorological data are also needed. Examples of instrument types appropriate for making these measurements are found in Table 11.4. (See also Chapter 2, "Radiation Instrumentation.")

For offsite dose assessment, simple equations must be developed that allow accurate calculation of integrated dose within 15 minutes of when data are received. A computer or desk-top calculator can be programmed with complex equations so that the insertion of required parameters is all that is

TABLE 11.4. Instruments for Emergency Radiological Measurements

Parameter	Instrument Types
Dose rate	Medium- to high-range ionization chamber
Surface count rate	Geiger-Mueller detectors Scintillators
Concentrations of radionuclides in air	Air-sampling device (air pump, vacuum pump) Analyzer: gas proportional counter or scintillation counter
Concentrations of radionuclides in effluents	Sampling devices (air, water, soil) Analyzer: gas proportional counter or scintillation counter
Meteorological conditions	Devices to determine wind speed and direction, temperature, and stability class

needed to run the program. Loss of electrical power must not affect the ability to make this calculation. The person responsible for assessment should be guided by the emergency plan on how to apply the assessment data to obtain projected doses.

B. Corrective Actions. Efforts must be made to reduce the likelihood that an accident will recur. In general, a thorough investigation is needed to identify areas that are weak and need strengthening. The results of the investigation should lead to appropriate corrective actions. If several alternative actions are possible, the action taken should be the one that incorporates, to the greatest extent possible, engineered safeguards rather than administrative guidelines.

C. Protective Actions. In an accident, all radiation doses should be kept as low as is reasonably achievable (ALARA) while the situation is brought under control. Limiting doses is best accomplished by limiting the release of materials through either engineered controls or manual actions. Because this is not always possible, protective actions should be developed to control the exposure of personnel and the public.

Examples of onsite actions that should be considered are:

1. providing protective clothing and respirators for use by emergency workers
2. sealing windows and doors and shutting off ventilation systems until conditions improve
3. removing personnel who are not contributing to the emergency response.

If personnel may need to be evacuated to offsite areas, routes and methods of evacuation should be planned and a destination upwind from any release should be identified. Provision must exist for transport vehicles, radiological surveys of personnel and vehicles, and offsite decontamination.

Emergency plans must also include ways of accounting for onsite personnel. Procedures should specify

1. personnel assembly points
2. the individual(s) responsible for accountability at each point
3. the individual to whom accountability status is reported
4. the individual responsible for notifying search-and-rescue teams.

As a general rule, the names of all missing persons should be determined within 30 minutes of the declaration of an emergency. All personnel remaining onsite should be continuously accounted for.

11.1.8 Facilities and Equipment

An emergency plan and the response based on it can be effective only if adequate facilities and equipment are available. For example, an offsite monitoring team would be useless if it did not have monitoring instruments that could measure in the range of emergency conditions or if it did not have communications equipment to report back the information gathered. The design of facilities and the types of equipment required for effective response depend largely upon the maximum credible accident and other postulated accidents. A variety of considerations in the design and selection of facilities and equipment for handling both small- and large-scale accidents is presented below. Judgment should dictate which considerations are appropriate for a given installation.

A. Emergency Control Centers. To facilitate the coordination, direction, and evaluation of the emergency response for site and general emergencies, one facility should be designated as the emergency control center (ECC). Because this area would be the hub of activity in an emergency, its location and design should be considered carefully. The ECC should have a low probability of being affected by an accident. If a postulated accident would result in high radiation levels in the ECC, its location should be changed.

Space is a primary requirement of the ECC. Adequate space must be allotted for each activity or group involved in the emergency response. Consideration must be given not only to the number of persons involved, but also to the space needed for chairs, tables, and monitoring and communications equipment. The assignment of space to groups is also important; groups that work together should not be on opposite sides of the room or across the hall from each other.

The onsite and offsite communications system in the ECC is another primary consideration. The system should be operational within 15 minutes after the activation of the ECC. The director of each emergency response organization must have at least one dedicated communications link between the organization and the ECC. The emergency director should have several open lines available for use.

The facility that is set aside as the ECC should be reserved for emergency use only. The emergency supplies kept there should be periodically inventoried and replenished as needed. Items that should be available in the ECC (depending on the scope of the postulated accidents) include:

1. the documented emergency plans, procedures, and checklists for the site
2. state and local emergency plans and procedures
3. emergency power
4. survey meters
5. air samplers
6. sample-counting equipment (unless adequate provisions are available for counting samples offsite)

7. personnel dosimeters for all the occupants
8. calibration sources
9. site and area maps marked with preselected monitoring points, locations of thermoluminescence dosimeters (TLDs), and environmental air sampling stations (useful maps are the U.S. Geological Survey 7-1/2-minute maps, which cover the plume exposure EPZ and are marked with cardinal polar coordinates and 22-1/2-degree sectors, with the first section splitting true north)
10. a board for posting emergency assignments and team designations
11. a board for posting up-to-date meteorological conditions and estimated doses at given distances from the release
12. as-built facility and building layouts
13. first aid kit and decontamination supplies
14. clock
15. writing materials and note pads
16. protective clothing
17. dose assessment equipment such as calculators
18. basic reference material
19. communications equipment (telephone, radio, etc.).

B. Medical Treatment Facility. Provisions must be made for either the installation's health personnel or a local hospital to care for contaminated individuals who are injured in an emergency. Information may be found in AR 40-13. Briefly, the following needs should be considered when a center for handling contaminated patients is being designed and equipped:

1. easy and immediate access
2. stretchers
3. first aid equipment and supplies
4. communication link
5. medical personnel trained in the handling of contaminated patients

6. operable, calibrated instruments for surveying contaminated patients
7. documented procedures for decontaminating patients
8. source of water and suitable decontaminants
9. provisions for the collection and disposal of solid and liquid waste.

C. Assembly Areas. Assembly areas where personnel gather when an alert is sounded should be able to accommodate the assigned number of persons. Consideration should be given to the adequcy of shielding, ventilation, rest rooms, communications equipment, and portable lighting for these areas.

D. Communications Equipment. Many types of communications equipment can be used during an emergency, including alarms, pageboy call systems, walkie-talkies, telephones, and two-way radios. The operation of each piece of equipment should be checked regularly and personnel should be trained to use the equipment.

Each communications link should have a back-up and an alternate power source. In addition, at least one communications system should provide uninterrupted service during a power failure.

Areas or groups that should be equipped with a communications system include:

1. the ECC, the emergency director, and directors of emergency response organizations
2. assembly areas and medical facilities
3. onsite monitoring teams
4. offsite monitoring teams
5. security personnel
6. the public (if applicable to postulated accidents).

The range of communications equipment used by monitoring teams and security personnel must be known. If the offsite monitoring team uses a two-way radio to communicate with the ECC, the radio must be able to transmit over the required distance.

In a general emergency, a communications system must be available to warn the affected public. Sirens mounted on telephone poles, or the local fire or police station, can serve this purpose. The public must know what action to take when alarms sound.

E. Monitoring Equipment. Onsite and offsite radiation-monitoring equipment must be capable of measuring the types and levels of radiation expected during a postulated accident and must be calibrated in the postulated accident range, using a source traceable to the National Bureau of Standards (see Chapter 2). For this reason, it is suggested that a number of portable instruments be dedicated to emergency response situations. These instruments should be checked routinely for operability and should be calibrated annually.

Many factors affect the choice of fixed and portable instruments for emergency response. The instruments must be capable of responding in extreme environmental conditions, such as high or low temperatures or humidity. Because many instruments do not operate in temperatures below -10°F, the manufacturer's performance specifications (which indicate the range of operability of an instrument) should be checked, and the instruments should be tested in the field during extreme weather conditions.

The accessibility of fixed instrumentation during postulated accidents should be assessed. If valuable data would be lost due to inaccessibility, remote readouts should be considered. A power failure may also render an instrument or its data inaccessible. If a particular instrument's data is necessary for accurately assessing the impact of an accident, provision should be made so that it will continue to function during a power failure.

Fixed air monitors can warn of airborne radiological hazards if they are designed to trip an alarm that will be heard or seen by site personnel. Therefore, these alarms should be placed at manned locations.

Records should be kept for each instrument that will be used in an emergency, documenting the type of radiation the instrument is designed to measure and the maximum and minimum radiation levels it can detect. The dates of and data from operational checks and calibrations should also be documented. A label indicating the date of the latest operational check and calibration and any conversion factors to be used in data interpretation should be placed on

each instrument. The storage location of all portable instruments and supplies for emergency response should be documented in the emergency procedures.

Kits used for onsite measurement and monitoring of radiation should be easily accessible for determining the initial accident conditions. The kits should contain a high-range dose rate meter, a contamination monitor, portable sampling devices, and light protective clothing. Electronic equipment should be tested periodically for operability, and the contents of the kit should be inventoried routinely. A breakable seal should be placed on each kit immediately after inventory so that any intrusion into the kit can be detected. An inventory should be taken promptly upon the discovery of a broken seal. A sample listing of emergency kit equipment is provided in Appendix C.

F. Aerial Monitoring. When an effluent release (the plume pathway) is being tracked, unfavorable meteorological conditions or the passage of the plume over inaccessible areas may hinder an accurate determination of the plume's location. In such cases, aerial surveillance using helicopters or fixed light-wing aircraft can contribute valuable information by providing survey data over a large area. Helicopters are best suited for emergency radiation surveys because of their maneuverability and slower flying speeds. A two-man crew (the pilot and someone to operate the radiation detection equipment) would generally be needed for such aircraft.

G. Dosimeters. Dosimeters that are designated for use only in emergencies should be available for each member of the emergency response team. These dosimeters must be capable of responding to the types and levels of radiation that would be present during postulated accidents. Pocket ionization chambers should be worn and checked frequently, especially by onsite and offsite monitoring teams. Each member of the emergency response organization should be assigned a film badge or TLD or both to record the dose received during the emergency.

H. Transportation Modes. Vehicles must be available to transport injured persons to either an onsite or an offsite medical facility. If an ambulance from a nearby hospital will be used, prior arrangements must be made for immediate service. Monitoring teams need vehicles for their exclusive use that can

carry monitoring equipment and emergency kits and handle any environmental or road conditions that may be encountered.

11.1.9 Offsite Agreements and Support

Offsite support can be invaluable in accident situations. Personnel at an installation cannot always perform all the tasks needed to respond to an emergency. Areas in which offsite support may be needed are fire fighting, health physics, security, and medical aid.

Advance agreements should be made with support organizations for their assistance. The agreements should specify the support to be provided and the conditions under which that support will be used.

11.1.10 Re-Entry and Recovery

During the period between the end of an emergency and restart of operations affected by the accident, imminent danger is not expected but the potential for higher-than-normal exposures may exist. The emergency plan should provide guidance on keeping these exposures to a minimum and ensuring that no recovery actions would place the installation back in an emergency situation.

Evacuated buildings must be re-entered with caution and only after a complete hazard assessment has been made and the emergency director has authorized re-entry. The only exceptions to these conditions may be for firefighting and search-and-rescue teams, whose activities must be supervised by the health physics staff.

The following topics relating to re-entry and recovery should be addressed in the emergency plan:

1. the conditions (e.g., exposure rates, radionuclide concentrations) under which rooms or buildings may be re-entered prior to their return to normal operation
2. the identification of personnel to direct re-entry and recovery
3. the assurance of proper communications to keep site personnel, response organizations, and DA and NRC personnel informed of progress in re-entry and recovery.

Section 11.2 MAINTAINING A STATE OF EMERGENCY PREPAREDNESS

Maintaining a state of emergency preparedness requires the effort of every individual within the installation. Each person needs to understand his or her responsibility and how it helps ensure the safety of the installation and its occupants. Emergency telephone numbers should be posted next to telephones, and diagrams of evacuation routes and lists of emergency signals with their meanings should be posted on bulletin boards or in hallways. Maintaining emergency preparedness includes:

1. training and retraining staff and emergency response personnel
2. conducting emergency drills
3. maintaining and inventorying emergency equipment, instruments, and supplies
4. reviewing and updating plans and procedures.

11.2.1 Training Staff and Emergency Response Personnel

All staff members and emergency response personnel must be familiar with the radiological emergency plan if it is to be effective. They should receive training in:

1. safety and accident control features specific to the facility to which they are assigned
2. the emergency signals (sirens, alarms), their meaning, and the expected response
3. the location of emergency assembly areas
4. the building layout, including emergency exits and evacuation routes
5. notification procedures and immediate actions if they discover or are involved in a radiation accident.

Personnel assigned emergency response duties require additional training in the proper execution of their duties. A representative list of persons or groups requiring this specialized training includes:

1. directors and coordinators of the plant emergency organization (see Table 11.3)
2. personnel responsible for radiological assessment
3. radiation-monitoring and survey teams
4. radiation protection personnel
5. maintenance teams
6. security personnel
7. search-and-rescue teams
8. firefighting squads
9. medical personnel
10. communications personnel
11. staff of state and local agencies and offsite support teams (if applicable).

Formal lesson plans should be drawn up for each training session, and the training program for each group should be documented. Each training program should give personnel an understanding of the emergency response plan and the role that each group plays in its implementation. The specific duties of each group and how these duties are to be performed (e.g., how to use equipment, whom to notify when, and how to treat a contaminated wound) should be included. Special precautions to observe in the performance of radiological emergency duties should also be included in the training program (see Appendix D). Whenever possible, practical hands-on operation of equipment and facilities should be included in the training program.

The quality of training depends to a large extent upon the quality of the instructors. A good instructor is professionally competent and has good communication skills. The instructor must also be thoroughly familiar with the emergency plan and each person's role in it. An effective training program may require the combined efforts of several individuals or organizations.

Retraining is important in maintaining a state of emergency preparedness. Because emergency duties are seldom performed, they are easy to forget. Formal training sessions should be held at least once a year.

Provision must be made for evaluating the ability of individuals to perform their emergency duties. The conditions, tasks, and standards of performance that form the basis for this evaluation should be documented. Attendance records and test scores from training sessions should also be documented.

11.2.2 Training Members of the News Media

When emergency planning includes offsite locations, training should be offered to individuals from the local news media. Newspersons should be trained in basic radiation protection practices and associated terminology. During an accident, one location should be designated as the media center, and all newspersons should be directed to that area upon arrival at the installation. The public relations spokesperson from the installation should be responsible for providing the media with up-to-date information, to help avoid conflicting stories and general confusion among the reporters and to help maintain credibility with the public.

11.2.3 Conducting Emergency Drills

Emergency plans should be tested annually through the use of emergency drills (AR 385-11). Drills jog memories, lead to the application of skills learned in training sessions, and keep interest in emergency response duties high. Drills also allow problem areas to be identified and corrected under controlled rather than accident conditions. In a full-scale drill, all onsite and offsite participants respond to a simulated severe accident. Smaller-scale drills involving specific response organizations should be held every 6 months.

11.2.4 Maintaining and Inventorying Emergency Equipment

To maintain a state of emergency preparedness, a schedule for maintaining equipment and supplies should be developed and followed. The inventory of kits and supplies should be checked periodically for completeness. This check should include operating and calibrating all instruments. The maintenance procedures should specify the corrective actions to be taken promptly when deficiencies are found during these checks.

11.2.5 Reviewing and Updating Plans and Procedures

When conditions change within an installation, emergency plans and procedures may need to be changed to meet the new conditions. The extent of the updating needed may range from changing a name on a call list to reassessing potential accidents if a new radiological function is defined. To ensure the adequacy and effectiveness of emergency preparedness, provisions should be made for a periodic review and update of the radiological emergency plan. A full-scale review should be conducted annually by a committee designated for this purpose in the emergency plan. This committee would ensure that:

1. the emergency plan and procedures are current
2. training sessions and drills have been conducted on schedule, test scores and drill deficiencies have been documented, and corrective actions have been taken
3. the emergency plan addresses the postulated accidents.

An individual or a committee should also be assigned to make necessary changes in call lists or equipment inventories as they occur. The name of the person or persons responsible for such changes should be documented in the emergency plan.

Section 11.3 NOTIFICATION AND REPORTING REQUIREMENTS

The DA criteria for defining radiation accidents are based on individual exposures, effluent releases, damage to property, and loss or theft of radioactive material and are given in Table 11.5. Both Army personnel and civilian licensing agencies must be notified when accidents that meet these criteria occur. Tables 11.6 and 11.7 list how soon notification is required for different accident levels, as set forth by DA (AR 385-40), NRC (10 CFR 20), and the Department of Transportation (DOT) (49 CFR 171). Other requirements for notification and for investigations and reports are given below for the three groups.

TABLE 11.5. DA Criteria for Defining Radiation Accidents (AR 385-40)

Type of Accident	Criteria
Individual Exposure	1. External exposure: Exposure greater than limits in 10 CFR 20 2. Internal exposure: Airborne concentrations in a restricted area, or ^{234}U, ^{235}U, ^{238}U concentrations greater than limits in 10 CFR 20, Appendix B, Table 1, Column 1 3. Fatality, lost-time injury, restricted-duty work
Effluent Releases	Greater than 500 times the limits in 10 CFR 20, Appendix B, Table II (averaged over 24 hours)
Damage to Property	1. Cost is $300.00 or more 2. Loss of facility operation for 1 day or more
Loss or Theft of Radioactive Material	Quantity that may result in substantial hazard to personnel in unrestricted areas

TABLE 11.6. NRC AND DA Notification Requirements for Accidents Involving Licensed Materials[a]

Notification	Individual Exposure	Release	Damage to Property
Immediate	Whole body (head, trunk, active blood-forming organs, lens of eye, gonads) ≥25 rem Skin ≥150 rem Extremities ≥375 rem	>5000 x amount listed in 10 CFR 20, Appendix B, Table II, averaged over 24 hours	>$200,000 Loss of ≥1 week of facility operation
Within 24 hours	Whole body (head, trunk, active blood-forming organs, lens of eye, gonads) ≥5 rem Skin ≥30 rem Extremities ≥75 rem	>500 x amount listed in 10 CFR 20, Appendix B, Table II, averaged over 24 hours	>$2,000 Loss of ≥1 day of facility operation

(a) Excerpted from 10 CFR 20 and AR 385-40.

TABLE 11.7. DOT and DA Notification Requirements for Accidents Involving Army Motor Vehicles Carrying Licensed Materials[a]

Notification	Individual Exposure	Damage to Property
As soon as practicable	Any event that presents a hazard to personnel at the site	>$50,000
	Fatality or lost-time injury	Fire, breakage, slippage, or suspected radioactive contamination

(a) Excerpted from 49 CFR 171 and AR 385-40.

11.3.1 Notification and Reporting Requirements: Army

The criteria indicating what constitutes radiation accidents are further subdivided into four DA classifications based on the degree of damage caused by the accident. These four general classifications are used for all Army accidents except aircraft mishaps.

1. Class A accident
   a. property damage, injury, or occupational illness costing $200,000 or more
   b. fatality as result of Army operations
   c. fatal injury of off-duty Army military personnel.
2. Class B accident
   a. property damage, injury, or occupational illness costing between $50,000 and $200,000.
3. Class C accident
   a. property damage costing between $300 and $50,000
   b. loss of one or more workdays due to injury or occupational illness.
4. Class D accident
   a. property damage less than $300
   b. one or more days of restricted work activity due to injury or occupational illness
   c. nonfatal case without loss of workdays.

Although immediate emergency actions and notification do not depend on these classifications, recording, reporting, and investigation requirements do.

In addition to the accident criteria for individual exposures specified in Table 11.5 and those described by Classes A, B, C, and D above, AR 40-14 defines three types of radiation overexposures to individuals. These classes are summarized in Table 11.8, and the reporting requirements are specified in Section B below.

A. Notification. The following Army personnel must be notified by telephone or electrical means immediately or within 24 hours of an accident (see Table 11.6) (this notification applies to Type III individual overexposures in Table 11.8):

1. the affected major Army commander or his representative
2. the licensee

TABLE 11.8. DA Criteria for Individual Radiation Overexposures (AR 40-14)

Body Part	Type I Overexposure	Type II Overexposure	Type III Overexposure
Whole body, head and trunk, active blood-forming organs, gonads, lens of eye	>400 mrem/mo[(a)] but <1.25 mrem/qtr	(b)	>5 rem/yr or >1.25 rem/qtr
Skin of whole body, forearms, cornea of eye	>3 rem/mo but <7.5 rem/qtr	(b)	>30 rem/yr or >7.5 rem/qtr
Hands and wrists, feet and ankles	>6 rem/mo but <18.75 rem/qtr	(b)	>75 rem/yr or >18.75 rem/qtr
Other organs (bone, thyroid, tissue, organ systems)	>1 rem/mo but <5 rem/qtr	(b)	>15 rem/yr or >5 rem/qtr

(a) mo = calendar month; qtr = calendar quarter; yr = calendar year.
(b) Dose rate exceeds the quarterly rate for a Type I overexposure but is less than the annual rate for a Type III overexposure.

3. HDQA (DAPE-HRS), AUTOVON 225-7291; DASG-PSP, AUTOVON 227-2796
4. HQ DARCOM (DRCSF-P), AUTOVON 284-9340
5. the Chief of Engineers (DAEN-FEZ-N), AUTOVON 354-5501, if the accident occurs at a reactor facility.

B. Reports and Investigations. The initial report must contain the following information:

"This is a Radiological Accident Report, RCS:DD-SD(AR)1168."

1. the date of the event
2. the radiation-producing device or source involved, including its national stock number, nomenclature, and radiation characteristics and parameters
3. a description of the event, including the cause, the name and social security number of each person exposed, estimated exposures and dose rates, contamination levels, facilities affected, and actions taken
4. any action taken to prevent a recurrence
5. recommendations on how to avoid similar accidents at other installations possessing similar material
6. a specific contact (name, address, telephone number)
7. a statement of when appropriate DA, NRC, and DOT offices were notified.

Class A, B, and C accidents must be documented and a report (DA Form 285) must be submitted to the U.S. Army Safety Center in Fort Rucker, Alabama, within 30 days of the accident. All Class A accidents require a formal board of investigation. This board is appointed by the commander to whom the radioactive materials license has been issued. Class B and C accidents are investigated by the local commander. Reports of these investigations should be forwarded through channels to HQDA (DAPE-HRS, DASG-PSP), Washington, DC 20310 and to Commander, DARCOM (DRCSF-P), 5001 Eisenhower Avenue, Alexandria, VA 22333, within 90 days of the accident. The requirements of the Privacy Act of 1974 must be taken into account whenever an individual is identified.

An informal investigation of Type I individual overexposures (see Table 11.8) is conducted by the immediate commander. The commander must

conduct a formal investigation of Type II and Type III overexposures and forward a report of the investigation and of the corrective actions through command channels to HQDA (DASG-PSP), Washington, DC 20310.

The investigation of a radiation accident can establish its cause and identify corrective and protective actions that will prevent the recurrence of the accident. The investigating individual or group should:

1. collect and preserve evidence
2. interview witnesses
3. prepare diagrams of the accident scene
4. re-enact the accident if appropriate.

When trying to establish the cause of an accident, the investigator(s) should consider possible defects in a component's basic design or construction. If a component is faulty, it should be identified in the investigative report by name, model number, manufacturer, and name-plate data. Other possible causes of an accident that should be considered are human error or misjudgment, incomplete or incorrect procedures, or the absence of procedures.

11.3.2 Notification and Reporting Requirements: NRC

Either NRC or an agreement state[(a)] licenses Army installations to use radioactive materials.

A. Notification. If the license is from NRC, the director of the appropriate NRC Inspection and Enforcement Regional Office (see 10 CFR 20, Appendix D) must be notified of an accident. If the license is from an agreement state, the director of the branch of state government issuing the license must be notified. Notification time shall be as described in Table 11.6.

B. Reports. A formal written report must be sent within 30 days of any accident to the appropriate NRC regional office listed in 10 CFR 20, Appendix B. A copy of this report should be submitted to the Director of Inspection and Enforcement, USNRC, Washington, DC 10555.

(a) An agreement state is any state with which NRC has entered into an effective agreement under Section 274 b. of the Atomic Energy Act of 1954, as amended (73 Stat. 689).

Reports of theft or loss of licensed material or of individual overexposures (Type III) should include the following:

1. a description of the licensed material involved, including the kind, quantity, and chemical and physical form
2. a description of the circumstances under which the loss, theft, or overexposure occurred
3. a statement of the disposition or probable disposition of the licensed material involved
4. quantitative radiation exposures to individuals and the extent of possible hazard to persons in unrestricted areas
5. actions that have been or will be taken to recover lost or stolen material
6. procedures or measures that have been or will be adopted to prevent a recurrence of the loss, theft, or overexposure.

After filing the written report, the licensee shall also report any substantive additional information on the accident within 30 days after the licensee learns of such information.

In reports filed with NRC, the names of individuals who may have been exposed to radiation shall be stated in a separate part of the report, including for each individual exposed the person's name, social security number, and date of birth, and an estimate of the individual's exposure. The requirements of the Privacy Act must be taken into account whenever an individual is identified.

11.3.3 Notification and Reporting Requirements: DOT

Each carrier must notify DOT at the earliest practicable moment after a transportation accident specified in Table 11.7. Notification should be given by telephone ((800)442-8802) and should include the following information:

1. the name and phone number of the individual reporting the accident
2. the name and address of the carrier represented by the individual reporting the accident

3. the date, time, location, and nature of the accident

4. the classification, name, and quantity of radioactive materials involved

5. the extent of injuries, if any, and whether a continuing danger to life exists at the accident scene.

A written report must be submitted to DOT in duplicate within 15 days of the accident.

REFERENCES

Brodsky, A. 1980. "Determining Industrial Hygiene Requirements for Installations Using Radioactive Materials." Health Phys. 38:1155-1171.

International Commission on Radiological Protection (ICRP). 1978. The Principles and General Procedures for Handling Emergency and Accidental Exposures of Workers. ICRP 28, Pergamon Press, Oxford.

Privacy Act, 1974. U.S. Code, Title 5, Section 552a.

U.S. Code of Federal Regulations. 1982. Title 10, Part 20, "Standards for Protection Against Radiation." U.S. Government Printing Office, Washington, D.C.

U.S. Code of Federal Regulations. 1982. Title 10, Part 30, "Rules of General Applicability to Domestic Licensing of Byproduct Material." U.S. Government Printing Office, Washington, D.C.

U.S. Code of Federal Regulations. 1980. Title 49, Part 171, "Hazardous Materials Regulations - General Information, Regulations and Definitions." U.S. Government Printing Office, Washington, D.C.

U.S. Department of the Army, Headquarters. Medical Support - Nuclear/Chemical Accidents and Incidents. AR 40-13, Washington, D.C.

U.S. Department of the Army, Headquarters. Safety - Accident Reporting and Records. AR 385-40, Washington, D.C.

U. S. Department of the Army, Headquarters. Safety - Ionizing Radiation Protection (Licensing, Control, Transportation, Disposal, and Radiation Safety). AR 385-11, Washington, D.C.

U.S. Department of the Army, Headquarters, Army Materiel Command. DARCOM Disaster Control Plans, Annex E, "Radiological Accident/Incident Control."

U.S. Department of the Army and Defense Logistics Agency. Medical Services - Control and Recording Procedures for Exposure to Ionizing Radiation and Radioactive Materials. AR 40-14, DLAR 1000.28, Washington, D.C.

U.S. Nuclear Regulatory Commission and Federal Emergency Management Agency. 1980. Criteria for Preparation and Evaluation of Radiological Emergency Response Plans. NUREG-0654, FEMA-REP-1, U.S. Government Printing Office, Washington, D.C.

APPENDIX A

EXAMPLES OF CHECKLISTS

A.1 Emergency Plan Checklist

A.2 Minor Spill Checklist

APPENDIX A

EXAMPLES OF CHECKLISTS

A.1 EMERGENCY PLAN CHECKLIST

Emergency Planning

___ Has one person or organization been assigned the responsibility for emergency planning?

___ Does this person or organization possess the authority to accomplish the task?

___ Have the maximum credible accident and several of the most probable accidents been determined?

Emergency Classification

___ Is the emergency classification system consistent with potential hazards at the installation?

___ Have the existing or imminent conditions for each class been defined?

___ Are definitions of radiation range continuous but distinct for each class (no gaps or overlap in definitions)?

Emergency Organizations

___ Have all emergency response organizations been identified?

___ Has each organization been assigned its emergency responsibilities?

___ Has each key person within the organizations been assigned a responsibility?

___ Have enough people been assigned responsibilities so that the emergency plan can be carried out completely and efficiently?

___ Do all individuals have sufficient training to carry out their responsibilities?

Emergency Facility and Equipment Identification

___ Are all emergency response facilities identified and fully described?

___ Have all emergency response resources and equipment been identified?

___ Have all onsite and nearsite impediments to the response been identified, along with realistic suggestions on ways to minimize their effects?

Emergency Plan Implementation

___ Do procedures exist for implementing the emergency plan?

___ Are personnel assignments and methods of implementation clear?

___ Do implementing procedures ensure that all organizations are manned at the "alert" stage?

___ Do procedures ensure that all support organizations will be notified promptly of emergency situations?

___ Are emergency action levels defined?

Emergency Response

___ Are all response procedures functional and easy to understand?

___ Is the installation capable of assessing all possible radiological conditions that may exist onsite and offsite as a result of its operations?

___ Have corrective actions to mitigate an accident been identified?

___ Are recommendations for protective action established?

___ Are they consistent with the recommendations of offsite agencies?

Emergency Facilities and Equipment

___ Have all emergency response facilities and areas been described in the plan?

___ Is the ECC expected to be habitable through most accident situations?

___ Have all tools, assessment equipment, protective equipment, and other support equipment used in emergency response been described in the plan or in a procedure referenced in the plan?

Agreements with Offsite Support Groups

___ Have all necessary agreements been made for offsite support and cooperation?

___ Are the agreements specific and the agencies reliable?

Re-entry and Recovery

___ Is re-entry of evacuated buildings controlled?

___ Have radiological conditions been established under which buildings may be re-entered for return to unrestricted use?

___ Are key positions in the recovery organization identified and have the responsibilities associated with those positions been assigned?

___ Is a communications system in place?

A.2 MINOR SPILL CHECKLIST

In the event of a minor spill of radioactive materials, the following checklist should be used.

Immediate Actions

___ Alert everyone in the immediate vicinity of the spill.

___ Have everyone leave the room and assemble in a nearby area such as a hallway. Allow no one to leave the area without a radiation survey. If the spilled material is highly toxic, evacuate the building to an assembly area. No re-entry should be attempted without health physics supervision.

___ Call for health physics assistance.

___ If the material does not present a hazard through toxicity or high dose rates, attempt to stop the leak and contain the contaminant with absorbent pads or other barriers. Try to minimize personnel contamination and exposure.

___ If large quantities of gaseous or highly volatile materials have been released, promptly shut down all heating, ventilation, and air conditioning operations to prevent the material from spreading.

___ Seal off the area with signs, rope, or locked doors until health physics assistance arrives.

Recovery of the Spill Area

___ Have health physics personnel supervise all recovery and re-entry activities.

___ Ensure that all persons involved in the accident or in recovery procedures are surveyed and decontaminated, if necessary, before release.

___ Try to determine the types and quantities of radioisotopes involved so that appropriate protection is used upon re-entry.

___ Establish a step-off pad at the entrance to the affected rooms.

___ Enter the room with appropriate protective clothing and devices, including dosimeters. Ensure that release of the material is halted and that cleanup can be performed without personnel receiving unacceptable doses (evaluate the radiological hazards).

___ Decontaminate the area, being careful not to spread contamination over an area larger than necessary.

___ Collect contaminated waste in plastic bags as it is generated, for later disposal.

___ Make a final survey of the room before it is released for use.

___ Have dosimeters processed promptly.

APPENDIX B

RESPONSE ACTIONS

B.1 Exposures to Individuals

B.2 Transportation Accidents

APPENDIX B

RESPONSE ACTIONS

B.1 EXPOSURES TO INDIVIDUALS

The magnitude of abnormal radiation exposures is not always apparent immediately following an accident. Radiation protection, medical, and administrative decisions will be based on a combination of all available data. However, the immediate care of an injured individual is of prime importance. Initially, any severe physical injuries (e.g., burns, cuts, or trauma) are likely to be more important than possible radiation injuries. Therefore, the extent of the injuries and the mobility of the patient should be assessed immediately, and first aid and lifesaving actions should be performed. Specific actions to be taken if contamination of a wound or the skin accompanies the physical injury are discussed below. (See also Chapter 7.)

In order to identify the response actions appropriate for individual radiation exposures, it is useful to define three categories: external exposure, internal contamination, and external contamination.

External Exposure

The level of action needed to respond to an external exposure depends on the magnitude of the dose received. The individual should be removed from the work environment and an accurate assessment of exposure should be made. Action and investigation levels are defined in AR 40-14. A summary of response actions to various doses received by an individual is outlined in Table 11.9, based on Publication 28 of the International Commission on Radiological Protection (1978).

An accurate dose estimation becomes more important as the dose gets higher and can be accomplished through a combination of clinical, biological, biochemical, and physical assessments of the exposed individual. The information provided by personal dosimeters, reconstruction of the event, and identification of radiation fields can be used to assess the dose. In the case of exposure to neutrons, activation products in or on the body (e.g., in the

TABLE 11.9. Summary of Response Actions for Individual External Exposure

Dose	Response Actions
5-10 rem	Administrative actions, investigation
	Physical dose measurements
10-25 rem	More detailed administrative investigation
	Assessment of possible biological consequences
	Physician brought in to assess the need for and the extent and nature of clinical, biological, or biochemical examinations
>25 rem	Same as above, plus an examination by the physician

blood, on the hair, or on metal objects such as coins or watch bands) can also aid in this assessment. Observable clinical symptoms such as nausea and vomiting would appear in approximately 10% of individuals exposed to 75 to 125 rem.

Priorities for treatment, and response actions for individuals subjected to whole-body exposures, are given in Table 11.10.

Internal Contamination

If an intake is suspected, first aid should be given immediately, the nature and degree of contamination should be determined, and therapy procedures should be started under the direction of a physician.

The initial indications for therapy include the first dose assessment and the results of nose blows and of monitoring for skin contamination, contaminated wounds, and, if appropriate, air and surface contamination. Examples of types of therapy to consider are: 1) isotopic dilution of an ingested radioactive substance by the administration of a stable isotope (e.g., administration of stable iodine, as sodium iodide or potassium iodide, to reduce the deposition of radioiodine in the thyroid gland); 2) acceleration of excretion through the administration of a laxative to minimize gastrointestional absorption; and 3) administration of irritants or expectorants to minimize respiratory absorption. Actions to be taken following a suspected internal contamination are presented in Table 11.11.

TABLE 11.10. Actions to be Taken Within Six Hours Following a Whole-Body Exposure

Medical Management

- Administer lifesaving treatment
- Check for external contamination
- Remove clothing and wash contaminated areas
- Give mild sedative for nausea and vomiting

Clinical Observation

- Collect dosimetric data
- Interrogate patient about accident and relay information to dosimetry team
- Make tentative prognosis based on above findings

Biological Investigations

- Take and keep urine samples
- Take blood samples for immediate cell counts, biochemical analysis, lymphocyte culture, and chromosomal analysis

Dosimetric Studies

- Process all personal dosimeters from exposed individual and bystanders
- Check installed recording equipment in vicinity of accident
- If neutron exposure is suspected, measure induced activity using coins the exposed person was carrying
- Make first assessment of likely type, quantity, and distribution of radiation, and inform physician
- Interrogate bystanders

Administrative Actions

- Perform detailed inquiry into the circumstances of the accident

---

(a) Excerpted from ICRP 28.

TABLE 11.11. Actions to be Taken Following Suspected Internal Contamination

Medical Management

Preliminary therapy (under the direction of a physician)
- isotopic dilution
- expectorants
- laxatives
- chelating agents

Biological Investigations

Take swabs from the nose or mouth
Perform whole-body count
Collect urine and fecal samples
Take blood sample

Dosimetric Studies

Confirm intake
Check installed air monitors
Make direct measurements using an external or wound probe and an organ scanner
Perform radiochemical assay of urine, fecal, and blood samples

Administrative Actions

Perform detailed inquiry into the circumstances of the accident

External Contamination

External contamination can involve both an external dose and internal contamination. First aid (including decontamination procedures) should be given immediately, and the dose received and the extent of contamination should be assessed promptly.

The individual should be decontaminated as effectively as possible before being taken to the hospital. Chapter 7 describes personnel decontamination procedures in detail. A few simple procedures are mentioned here. Skin areas are decontaminated by washing them with soap and large amounts of water. The contaminated individual can often do this. Measurements of residual contamination should be taken after each washing. However, this treatment should cease before skin abrasions appear. The eyes, nose, and mouth can be decontaminated by flushing them with large quantities of water. Contaminated wounds should immediately be washed with large quantities of water, and bleeding should be encouraged. The use of a chelating agent is recommended. All of the procedures except for the washing of skin areas require the supervision of medical personnel (see Chapter 7).

B.2 TRANSPORTATION ACCIDENTS

Accidents that occur during the shipping of radioactive materials may require the involvement of state and local authorities, and/or the DOT. Appropriate responses to the accident include the following actions:

1. Administer first aid to seriously injured persons and summon a rescue squad.
2. Confine contamination to the local area; an exclusion area may be established.
3. Locate people along the shipping route who may have been exposed or contaminated.

Federal interagency radiological assistance can be obtained by calling the Joint Nuclear Accident Coordinating Center at Kirtland Air Force Base (Commercial (505)264-8279 or AUTOVON 964-8279).

The nearest Army facility may also be called upon for assistance. Table 4-1 of AR 385-11 lists Army addresses and emergency telephone numbers.

APPENDIX C

EXAMPLE LISTING OF EMERGENCY KIT EQUIPMENT

APPENDIX C

EXAMPLE LISTING OF EMERGENCY KIT EQUIPMENT

Date checked_____

Checked by_____

Items	Quantity	Box 1	Box 2
Protective clothing			
Coveralls			
Neoprene gloves			
Disposable gloves			
Head covers			
Shoe covers			
Goggles			
Respirators			
Respirator cartridges			
Chemical			
Particulate			
Masking tape			
Posting equipment			
Radiation rope			
Radiation signs			
Radiation labels			
Radiation tape			
Masking tape			
Twine			

Items	Quantity	Box 1	Box 2
Tools			
Scissors			
Tongs (46 cm)			
Extension cord			
Channel-lock pliers			
Screwdriver			
Radar light			
Knife			
Surveying and sampling supplies			
Cotton swabs			
Disposable bottles			
Large plastic bottles			
Scintillation vials			
Air-sampling filters			
Air-sampling cartridges			
Smears and smear holders			
Tweezers			
Plastic bags			
Large			
Small			
Decontamination aids			
Detergent			
Cleanser			
Gauze pads			

Items	Quantity	Box 1	Box 2
Miscellaneous			
Adhesive tape			
Pencils			
Notepads			
Butcher paper			
Stopwatch			
Extra batteries			
Readily Available Equipment			
Survey meters			
Ionization chamber			
Geiger-Mueller counter			
Air samplers			
Alpha detector			
Fast- and slow-neutron meters			
High-range pocket dosimeters			
Spare film badges			
Small fire extinguisher			
Portable power source			
First aid kit			

APPENDIX D

EXAMPLES OF EMERGENCY ACTIONS

D.1 Ambulance or Rescue Squad Personnel

D.2 Hospital Emergency Room Personnel

D.3 Firemen

APPENDIX D

EXAMPLES OF EMERGENCY ACTIONS

D.1 AMBULANCE OR RESCUE SQUAD PERSONNEL

Guidelines for handling patients contaminated with radioactive materials:

1. Give lifesaving emergency assistance if needed.[a]
2. If a health physicist is immediately available, have him or her ride with the patient in the transport vehicle.
3. Cover the stretcher and pillow with an open blanket; wrap the patient in the blanket to limit the spread of contamination.
4. Call the appropriate hospital by radio or telephone and provide available information.
5. Save all materials suspected of being radioactively contaminated in plastic bags or containers labelled with patient's name, date, and time.
6. Ensure that rescue squad personnel and equipment are monitored upon arrival at the hospital.

D.2 HOSPITAL EMERGENCY ROOM PERSONNEL

Upon notification of the imminent arrival of a contaminated patient, the following actions should be taken:

1. Notify responsible staff physician, hospital administrator, and health physicist.

(a) Note: Medical treatment takes precedence over personnel decontamination and/or contamination control.

2. Take precautions to prevent the spread of contamination:
   a. Prepare a separate space, using an isolation room or cubicle if available.
   b. Cover the floor with absorbent paper.
   c. Mark and close off the area.
   d. Prepare to shut off air circulation system, if dust is involved.
3. Obtain appropriate survey meter.
4. Put on protective clothing.

Upon arrival of the patient:

1. If patient is seriously injured, give emergency lifesaving assistance immediately.
2. Have health physicist check patient for contamination using survey meter. Record patient's name, date, time, location and extent of contamination, and radiation measurements.
3. If external contamination is involved, save all clothing and bedding from ambulance, all metal objects (jewelry, belt buckles), and all blood, urine, stool, and vomitus, and label with patient's name, date, and time. Store in plastic bags or containers marked "Radioactive - Do Not Discard."
4. Begin decontamination procedures (if patient's medical status permits) by cleansing and scrubbing the area of highest contamination first, using soap and warm water; showering may be necessary. Resurvey and record measurement after each washing or showering. If a wound is involved, use self-adhering disposable surgical drape to cover it, then cleanse neighboring skin surfaces and seal with surgical drape. Remove the wound covering and irrigate the wound with sterile water, catching the water in a basin marked "Radioactive - Do Not Discard."
5. Save physicians', nurses', and attendants' scrub or protective clothing. Follow monitoring and decontamination procedures.

D.3 FIREMEN

Special precautions must be taken in fighting a fire involving radioactive materials:

1. Identify and isolate the hazard.
2. Contact a health physicist for guidance and assistance.
3. Stay upwind from the fire.
4. Wear self-contained breathing apparatus and full protective clothing.
5. Limit time spent in hazard area to shortest possible time.
6. Avoid contact with leaking or damaged packages.
7. Fight fire from as far away as possible.
8. Move undamaged packages out of the fire zone if this can be done with no risk.

Additional information may be obtained from the National Fire Protection Association (NFPA 801-Recommended Fire Protection Practice for Facilities Handling Radioactive Materials).

CHAPTER 12. TRAINING

TABLES

CHAPTER 12. TRAINING

The Radiation Protection Officer (RPO) is responsible for conveying to all staff members policies and procedures relating to radiation safety. The extent and breadth of the training needed varies significantly with job requirements and responsibilities. For a clerk, a brief description of the working environment, an explanation of protective measures to be taken in case of an emergency, and an assurance of personal safety may be sufficient. For a handler of radioactive waste, extensive formal training is required.

Information on the radiation safety policy should be presented during the new staff member's orientation. At that time, a general introduction to the radiation hazards associated with the work should be given. Radiation hazards and related safety programs should be presented not as unique or special entities, but rather as part of the overall program for occupational health. Written material on these topics can be an invaluable resource for distribution to new employees.

This chapter describes the training that should be presented to radiation workers, women of reproductive capacity, users of respirators, managers, and radiation protection personnel.

Section 12.1 TRAINING FOR RADIATION WORKERS

The term "radiation worker" is synonymous with the term "occupationally exposed individual." A radiation worker is an individual whose work is performed in a radiation area or a controlled area and who might be exposed to more than 5% of the basic radiation protection standard listed in Chapter 3, Table 3.2, of this manual (see also AR 40-14) as a result of duties in these areas. Radiation worker training should be extended to all individuals who work in radiation areas or controlled areas even if they do not work directly with radioactive material. For example, fire fighters, security forces, emergency response personnel, janitors, and night guards who may need to enter a radiation or controlled area during the course of their work should receive

radiation training, as should those assigned to work full time in these areas. Required instruction for workers is detailed in 10 CFR 19.12.

12.1.1 Frequency of Training

Individuals should receive training before entering or beginning work in a radiation or controlled area. They should be retrained annually or whenever policies or procedures are changed.

12.1.2 Course Content

The training program should include the subjects listed in Table 12.1 and discussed below. The topics emphasized will vary with the needs of each individual or group being trained. Each individual's work assignment and the standing operating procedures (SOPs) covering the assignment should be carefully reviewed to determine the scope of training needed. Appropriate reference documents covering essential facts, requirements, regulations, procedures, and plant organization should be given to each individual.

A. Radiation Biology and the Risk from Occupational Exposure. Persons who work in or near radiation and controlled areas or make decisions about work in those areas should be taught enough about radiation effects to appreciate the importance of keeping exposures as low as is reasonably achievable (ALARA). These individuals should be informed of the level of radiation dose anticipated in their work area and the risk associated with such a dose level. Appropriate topics could include dose-effect relationships for internal and external radiation and the collective-dose concept of risk (individual and group) as it applies to the ALARA philosophy.

B. Radiation and Radioactive Material. Types of radiation and their characteristics should be discussed to the extent necessary to explain the nature of the material people work with. Types and forms of radioactive material should be detailed so that staff members understand proper control procedures. Sources and origins of radioactive material and radiation onsite should be identified, as should the signs and labels used to mark this material.

TABLE 12.1. Appropriate Subjects for a Radiation Protection Training Program

1. Radiation Biology and the Risk from Occupational Exposure
    a. Dose-effect relationship
        (1) External radiation
        (2) Internal radiation
    b. Collective-dose concept
        (1) Group total man-rem risk
        (2) Individual dose risk

2. Radiation and Radioactive Material
    a. Types of radiation and their characteristics
    b. Types and forms of radioactive materials
    c. Sources (origins) of radioactive materials and radiations onsite
    d. Source identification

3. Measurement and Control of Radiation Exposure and Radioactive Material
    a. Dosimetry
    b. Maximization of distance between people and radiation sources
    c. Shielding
    d. Detection and control of contamination, and decontamination
    e. Radiation measurement and survey instruments
    f. Area and air monitoring
    g. Personnel monitoring
        (1) Internal
        (2) External

4. Radiation Protection Program
    a. Radiation protection standards, guides, and limits
    b. ALARA program
    c. Responsibilities of individuals
    d. Radiation areas at the site
    e. Signs and labels
    f. Control of radiation areas
    g. Investigation and reporting of abnormal exposures
    h. Radiation surveys--purpose and methods
    i. Protective apparel
    j. Respirators and their use
    k. Rules and procedures, including standing operating procedures
    l. Professional guidance and assistance
    m. Control and removal of contamination and contaminated equipment

5. Emergency Preparedness
    a. Plant safety and accident control features
    b. Signals and alarms
    c. Evacuation routes and procedures
    d. Assembly points
    e. Communications
    f. Emergency equipment
    g. First aid and treatment of contaminated wounds

C. Measurement and Control of Radiation Exposure and Radioactive Material. Each radiation worker should be informed that radiation and radioactive materials can be measured at levels significantly below radiation protection standards and controlled by means of suitable design and procedural techniques. Radiation workers should understand the elements of radiation measurement and control well enough to participate in an effective radiation protection program consistent with the ALARA philosophy. Emphasis should be on 1) the sources of radiation, 2) contamination control, 3) the use of time, distance, and shielding to reduce doses, 4) SOPs, and 5) the proper use of dosimeters. The importance of administrative and engineered controls and the performance of work in accordance with carefully planned procedures should be stressed.

D. Radiation Protection Program. Personnel should understand the nature and scope of the radiation protection program, including pertinent portions of regulations, site rules for radiation protection, and safe operating procedures. Emphasis should be placed on the ALARA philosophy, its objectives, and its implementation within the framework of the tasks to be performed. The responsibility of the radiation protection staff in implementing ALARA goals, and the responsibilities of the individual staff member within the ALARA program, should be understood.

At the completion of the training program, radiation workers should understand that personnel outside radiation and controlled areas should not be significantly affected by activities in these areas that involve radioactive materials or radiation. The meaning and importance of posted instructions, including radiation warning signs and tags, and the importance of following instructions should also be understood.

E. Emergency Preparedness. Staff members should know the appropriate response to alarms and signals. They should be familiar with the details of emergency procedures and preparations so they will know what is expected of them and from whom they can expect guidance in an emergency. They should know the locations of emergency facilities and equipment as well as emergency escape routes and safe assembly points. Preparations for possible emergencies should be emphasized; such emergencies should include accidents involving

severe personal contamination, contaminated wounds, and localized fires in radiation and controlled areas.

12.1.3 Use of Mockup Facilities

The use of equipment or facility mockups allows individuals to practice procedures in a realistic setting before they perform the procedures using radioactive materials or enter areas where a potential for exposure to radioactive contamination exists. This type of training is especially valuable for repair and maintenance tasks that could result in high doses to personnel in relatively short periods of time. Another valuable application is in research laboratories where radioisotopes are used.

12.1.4 Evaluation of Trainee Performance

Each radiation worker's knowledge, competency, and understanding of the radiation safety aspects of specific jobs should be evaluated. The evaluation may consist of only a written or oral test, but should, in most cases, include a written test, an oral test, and a "practical" or on-the-job performance test. The questions asked and the responses given in all examinations should be documented. Requalification testing should be conducted in conjunction with refresher training.

High test grades (i.e., 80% or higher) should be required because each person's training covers radiation protection information relevant to the person's needs and safety in the work environment. Radiation workers should be reinstructed and retested in any areas in which their knowledge is shown to be deficient.

Tests should cover all the information presented during training but should emphasize the day-to-day radiation protection practices relevant to each person's job. As experience is gained, test questions should reflect the radiation protection problems actually experienced onsite.

Practical or on-the-job tests should stress knowledge and proper job performance. A person may know what to do but be unable to do it promptly when faced with a situation demanding immediate and effective action. In preparing a test, consideration should be given to individual job responsibilities, training received, and radiation protection experience.

Tests should be designed to:

1. measure the person's ability to recognize and cope with radiation hazards that may be encountered on the job
2. stress preparedness for work in radiation and controlled areas
3. assess the individual's knowledge of and attitude toward his or her rights and obligations regarding radiation protection
4. assess the individual's understanding of control procedures.

12.1.5 Documentation of Training

Records that describe the content of training courses, such as course outlines, syllabuses, brochures, video tapes, texts, and tests, should be maintained. These records serve as a basis for determining the depth and scope of training given in each subject area. Trainee-specific training records, which provide a complete history of each person's training experiences, should also be maintained. A complete description of information to be included in the training records is given in Chapter 13, "Recordkeeping."

A staff member who has been trained at one site and is later to be employed at a different site should receive a statement of training received. This statement will allow the person responsible for training at the second site to take the staff member's previous training into account and thereby avoid needless repetition of training. The statement should clearly and explicitly describe all training received and should identify non-plant-specific training segments that may be applicable to work in the new position.

Section 12.2 INSTRUCTION TO WOMEN OF REPRODUCTIVE CAPACITY

A special situation arises when an occupationally exposed woman is pregnant. Exposure of the woman's abdomen to penetrating radiation from either external or internal sources would also expose the embryo or fetus. A number of studies have indicated that the embryo or fetus is more radiosensitive than

an adult, particularly during the first 3 months after conception, when a woman may be unaware of her pregnancy.

12.2.1 Recommended Prenatal Occupational Exposure Limit

The National Council on Radiation Protection and Measurements (NCRP) recommends in its Publication 53 (1977) that, because the unborn are more sensitive to radiation than adults, their radiation dose from occupational exposure of the mother should not exceed 0.5 rem. The International Commission on Radiological Protection (ICRP) recommends in its Publication 9 (1965) that the occupational radiation exposure of all women of reproductive capacity be received gradually, in small increments, so that an unborn baby would be unlikely to receive more than 0.5 rem in the first 2 months after conception, when a woman may not be aware that she is pregnant.

12.2.2 Requirements

All individuals who work in a restricted area must be instructed as to the risks associated with radiation exposure (Nuclear Regulatory Commission (NRC) Regulatory Guide 8.13 (1975)). This instruction should include information on the risks to the unborn. Women should be encouraged to inform the RPO of a pregnancy. Every effort should be made to limit the dose to an embryo-fetus to 0.5 rem during the entire gestation period. The mother's exposure should be as uniformly distributed over time as is practicable.

The establishment of differential occupational exposure limits for men and women can raise a number of social and legal questions. All alternatives should be considered before the situation arises. Options include the following:

1. The dose to the unborn child can and should be reduced by a) decreasing the time the woman spends in radiation areas, and/or b) increasing the distance between the woman and the source of radiation, and/or c) shielding the abdominal area (the use of lead aprons could be considered).
2. The woman can be reassigned to an area or job involving less radiation exposure.
3. The woman can be reassigned to a nonradiation position.

All available options should be discussed with the expectant mother. It is important that a decision be reached quickly, as the unborn child is most radiosensitive during the first 3 months of pregnancy.

12.2.3 Rationale for Limit

The radiosensitivity of cells (their susceptibility to damage by radiation) is directly related to their degree of differentiation, that is, to the extent to which they have developed distinct and identifiable functions. Kidney cells, for example, have a function different from that of cells of the eye. Because most cell differentiation takes place in newly forming and growing beings, embryos are more radiosensitive than fetuses, fetuses more radiosensitive than children, and children more radiosensitive than adults. This principle has long been a factor in the development of radiation protection standards, as exemplified by the difference in the exposure limits for minors and adults: the occupational radiation exposure of anyone under the age of 18 cannot exceed 10% of the limits for adult workers.

The development of a baby is usually divided into three stages: ovum, embryo, and fetus. An ovum becomes an embryo about 7 days after fertilization; the embryo stage lasts approximately 8 weeks; and the fetal stage is the time remaining until birth. The particular effect of radiation, and its severity, depend on the stage of development at which exposure occurs. An unborn child is more sensitive to radiation during the embryonic stage than in the earlier or later stages of development. During this period, the organs are being formed and the cellular organization of the embryo is changing rapidly. Cells become specialized and start processes leading to the development, in a fixed sequence, of specific tissues. Consequently, the effect of radiation varies from day to day, and different degrees and kinds of organ malformations are produced depending on exactly when the exposure occurs.

During the earlier or ovum stage, relatively few cells are present, and the most common effect of exposure to radiation is chromosomal injury leading to cell death. During the later or fetal period, most organs have already been formed, and malformations from radiation exposure are less common and

less severe. The major radiation effect during this period is reduced growth, which may persist throughout life.

The genetic and cancer risks per unit of radiation dose from in-utero exposure also exceed those from adult exposure. The undeveloped ovum cells in the female fetus are actively dividing and are nearly as sensitive as the male fetus's immature sperm cells. The most sensitive period for genetic damage in both sexes is probably the last 6 months before birth.

The leukemia risk from in-utero exposure has been estimated as being 10 times greater than that for adults who get the same dose. The follow-up period for solid tumors, which have a longer latency period than leukemia, has probably not been long enough to allow a good estimate of the total risk for other cancers caused by in-utero exposures. The absolute risk of getting fatal cancer, other than leukemia, in the first 10 years of life from in-utero exposure, however, has been estimated as five times the risk that an adult has of getting cancer within 10 years of receiving the same exposure. For all of these reasons, the occupational radiation exposure of pregnant women should be limited.

Section 12.3 INSTRUCTION IN THE USE OF RESPIRATORS

Training in the use of respirators should be given by a qualified and experienced instructor, such as a health physicist, industrial hygienist, or safety engineer. The instructor must have a thorough knowledge of the application and use of respirators and of the hazards associated with radioactive airborne contaminants. He or she also must have had considerable experience in the practical selection and use of respirators for protection against radioactive airborne contaminants.

12.3.1 Extent of Training

The instructor should develop an adequate training program based on the hazards that may be encountered and the types of respirators to be worn. Training must be given not only to the persons who will perform work using the respirators but also to those who will direct the work. Especially where

respirators are used only occasionally, staff members should be retrained often enough so that a high degree of proficiency is retained when respiratory equipment is actually used.

12.3.2 Contents of Training Program

Training in the use of any respirator must cover at least the following topics:

1. the nature of the airborne contaminants against which the wearer is to be protected, including their physical properties, maximum permissible concentrations, physiological action, toxicity, and means of detection
2. the construction, operating principles, and limitations of the respirator and why the respirator is the proper type for the particular purpose
3. the reasons for using the respirator and why more positive control of airborne contamination is not immediately feasible, including recognition that every reasonable effort is being made to reduce or eliminate the need for respirators
4. procedures for ensuring that the respirator is in proper working condition
5. how to fit the respirator properly and how to check the adequacy of the fit
6. the proper use and maintenance of the respirator
7. application of available cartridges and canisters for air-purifying respirators
8. what emergency action to take if the respirator malfunctions
9. radiation and contamination hazards, and other protective equipment that may be used with respirators
10. classroom and field training in recognizing and coping with emergency situations
11. other special training as needed for special purposes.

12.3.3 Drills

Training should include actual use of respirators under simulated conditions of exposure so that the wearers develop a sense of confidence in their ability to use the devices properly. A qualified observer should review with the trainees their performance in these drills.

Section 12.4 TRAINING FOR MANAGERS

Managers need to be knowledgeable in all radiation safety policies and procedures and to understand the ALARA philosophy. They should know who the members of the radiation protection staff are and how to contact them.

12.4.1 Frequency of Training

Managers should be offered training when they move into a position which requires that they oversee work with radioactive materials. This training can often be done on a one-to-one basis. Retraining should be provided whenever a change in policy is made. A presentation at a regularly scheduled staff meeting is a convenient way to provide retraining.

12.4.2 Contents of Training Program

Training for managers should include the following topics:

1. basic radiation safety and radiation biology; sufficient detail should be provided to allow an understanding of the ALARA program
2. site-specific radiation program
3. responsibility of the manager
4. responsibility of staff members
5. emergency preparedness.

Section 12.5 TRAINING FOR THE RADIATION PROTECTION STAFF

The responsibilities of the RPO and the radiation protection staff were detailed in Chapter 3. Members of the radiation protection staff need training that will prepare them to meet those responsibilities and to maintain proficiency in their duties. Contact DARCOM Headquarters for assistance in identifying appropriate short courses.

REFERENCES

International Commission on Radiological Protection (ICRP). 1965. Recommendations of the International Commission on Radiological Protection. ICRP 9, Pergamon Press, Oxford.

National Council on Radiation Protection and Measurements (NCRP). 1977. Radiation Dose Limit for Embryo and Fetus in Occupationally Exposed Women. NCRP 53, Washington, D.C.

U.S. Code of Federal Regulations. 1982. Title 10, Part 19, "Notices, Instructions and Reports to Workers; Inspections." U.S. Government Printing Office, Washington, D.C.

U.S. Department of the Army and Defense Logistics Agency. Medical Services - Control and Recording Procedures for Exposure to Ionizing Radiation and Radioactive Materials. AR 40-14, Washington, D.C.

U.S. Nuclear Regulatory Commission (NRC). 1975. "Instruction Concerning Prenatal Radiation Exposure." Regulatory Guide 8.13, Washington, D.C.

CHAPTER 13. RECORDKEEPING

TABLES

CHAPTER 13. RECORDKEEPING

Good recordkeeping is essential in the radiation work environment. Accurate records and a filing system that incorporates extensive cross-referencing can help the Radiation Protection Officer (RPO) and the installation commander achieve the following:

1. plan an individual's occupational exposure, keeping in mind the ALARA philosophy (maintaining radiation exposures as low as is reasonably achievable)
2. demonstrate good management practices in the handling of radioactive sources
3. demonstrate compliance with government regulations and the site's Nuclear Regulatory Commission (NRC) license
4. evaluate the effectiveness of the radiation protection and quality assurance programs
5. trace the cause of a trend of elevated doses
6. document, for both legal and medical purposes, the exact conditions under which an individual received a particular radiation dose (i.e., what the radiation source was, its activity or probable concentration, and when and how the individual was exposed).

This chapter describes the content and form of the radiation work records that must be maintained in accordance with the requirements of the Department of the Army (DA) and the following parts of the U.S. Code of Federal Regulations: Title 10, Parts 19 and 20, and Title 29, Parts 570.57 and 1910.96. A method of organizing these records into a filing system that would provide easy access to all records pertaining to an individual, a specific project, a radioactive source, a radiation work area, or a particular radiation-measuring instrument is also described. The chapter closes with a section on retention and storage of records.

Section 13.1 RADIATION RECORDS FILES

A well-managed radiation protection program requires a substantial number of records. Many of these records have been described in previous chapters. In this section, a summary of the required records will be provided. For the purpose of this manual, the records have been organized into the following series of files:

1. personnel file
2. radiation protection program file
3. project file
4. radiation work area file
5. instrumentation and dosimeters file
6. radioactive-material inventory file
7. waste management file
8. transportation file
9. accidents/incidents file
10. training file
11. quality assurance file.

A system for cross-referencing these files is provided in the next section. A records filing system for radiation safety files is also given in AR 340-18-6. Each RPO should evaluate records requirements to determine what kind of filing system is most appropriate.

Reference will be made throughout this chapter to "suspense" files. These are files used for procedures that are repeated regularly (e.g., weekly, monthly, quarterly, or yearly). Data sheets for the particular procedure are filed under the week, month, quarter, or year in which the procedure must be performed next. Suspense files can take the form of card catalogs, spiral notebooks, or file folders, and are appropriate for scheduling routine procedures such as leak tests of sealed sources, contamination surveys of radiation areas, instrument calibration tests, training and retraining sessions, and bioassays.

13.1.1 Personnel File

Complete and up-to-date personnel files provide a means of 1) assessing a radiation worker's training needs for specific projects or job changes, and 2) tracking the history of the individual's exposures and of any doses received. Occupational exposure records must be kept as part of each individual's health record or civilian employee medical file. Each personnel file must include a signed Privacy Act statement (AR 40-14).

A. Identification of the Individual. An individual's social security number should be used for identification on all records. If another number is used to identify the individual, this number should be cross-referenced to the social security number. If an individual who may work with radiation does not have a social security number, he or she should be instructed to get one. The birth date and sex of the individual should also appear on all personnel records as another means of identification. In this chapter, "identification of the individual" will mean the person's name, social security number, birth date, and sex.

B. Training Records. Participation by a radiation worker in formal and on-the-job training sessions should be documented to indicate the individual's qualification to perform radiation-related tasks. The training records should include:

1. identification of the individual
2. title and date of the training program
3. identification of the instructor and training location
4. a performance rating for each segment of training or each training program satisfactorily completed: a numerical or letter grade and/or a written evaluation.

A suggested format for training records is shown in Appendix A.

C. Project/Task Listing. To facilitate tracing an individual's exposure history at a given installation, a listing of all the projects or tasks on which the individual has worked should be included in his or her personnel file. A useful concept is the assignment of a key word descriptor to each project. Key word descriptors are one- or two-word descriptions of the focus of a

project, for example, weapons testing, gaseous effluents, radioimmunoassay. They can be used to locate all other projects of the same type for either onsite or intersite comparisons. A record sheet for listing projects and tasks would include:

1. identification of the individual
2. title and number of the project or task
3. key word descriptor for the project or task
4. the dates on which the individual began and ended participation in the project or task
5. standing operating procedures (SOPs) for the project or task.

A sample record sheet is shown in Appendix A.

D. External-Exposure Records. Two Department of Defense (DD) forms are used to record an individual's occupational radiation exposure history: DD Form 1952 and DD Form 1141. (Both forms are reproduced in Appendix B. See AR 40-14 for details on the information summarized here.) Department of Defense Form 1952, "Dosimeter Application and Record of Occupational Radiation Exposure," identifies the individual's employment status, gives dosimetry information for the individual's current job (e.g., the type of exposure involved and the dosimeters and bioassays required in connection with the work), and lists the names and addresses of previous employers for whom the individual worked with radiation, with the dates of employment. A new DD Form 1952 is initiated each time the individual is reassigned, and the previous exposure history is transferred to the new form.

Department of Defense Form 1141, "Record of Occupational Exposure to Ionizing Radiation," includes the individual's identity, a summary of exposures from previous jobs, and a month-by-month record of the individual's dose from the current assignment and accumulated lifetime dose from exposures to the whole body or skin of the whole body. The installation or location at which each exposure occurred is also noted on the form.[a] A separate DD Form 1141

(a) The inclusion of the title and number of the project on which the individual received each monthly exposure would facilitate cross-referencing of this information with that in other files.

is kept to record exposures to parts of the body other than the whole body or skin of the whole body (e.g., the thyroid, head and neck, or fingers). An alternative to the use of DD Form 1141 is the use of the automated dosimetry records prepared by the Army's Central Dosimetry Record Repository. Whichever record is used, it should include exposures received by the individual from outside (non-Army) work and from medical sources.

Department of Defense Form 1952 is kept in the individual's health record or (for civilian employees) medical file. Department of Defense Form 1141 (or the automated dosimetry records) can be kept either in the individual's personnel file or in his or her health record or medical file. If DD Form 1141 is kept in the personnel file, a chargeout record noting the location of the form must be placed in the health record or medical file.

E. Internal-Exposure Records. Internal-exposure records can include bioassay data, the interpretation of bioassay data, whole-body-counter records, and airborne-radioactivity measurements. All internal-exposure records can be maintained either in the individual's personnel file or in the health record or medical file. If they are kept in the personnel file, a chargeout record noting their location must be placed in the health record or medical file.

(1) Records of Bioassay Data. An individual's internal radiation exposure is determined from bioassay studies. Records of these studies should include the following information (American National Standards Institute (ANSI) Standard N13.6-1972):

1. identification of the individual
2. purpose of the sample and, if applicable, date of suspected intake, work area, and project number and title
3. collection period for the sample and date submitted
4. type of sample and size of aliquot
5. type of radioactivity (e.g., alpha, beta)
6. gross and net activity observed and counting time
7. identity of radionuclide, when required

8. cross-reference to calibration and control data and confidence limits (in the instrumentation and dosimeters file)

9. cross-reference to identity and efficiency of analysis equipment and radiochemical analysis procedure (in the instrumentation and dosimeters file)

10. identification of the laboratory technician(s) performing the analysis.

(2) Records of Bioassay Interpretation. In addition to items 1 through 10 above, records relating to interpretation of the data from a bioassay study should be kept and should include:

1. a listing of the bioassay data used in the interpretation, and the identity of the radionuclide

2. reference to the method of interpretation

3. assumptions used in arriving at the conclusion, including the known or assumed date of exposure

4. conclusion as to the magnitude and location of the body burden, expressed in microcuries of the specific radionuclide

5. identification of the individual making the conclusion.

(3) Whole-Body-Counter Data. Whole-body-counter data provide an assessment of internally deposited radionuclides. Records of an individual's whole-body count should include:

1. identification of the individual

2. date, time, and purpose of the count and, if applicable, date and time of suspected intake

3. quantitative data (e.g., length and type of count, counts per channel, keV per channel, energy range over which counts were made)

4. cross-reference to procedure, calibration factors, periodic background and resolution checks, and confidence levels (in the instrumentation and dosimeters file)

5. description of or reference to calculational procedure

6. identity and location of the radionuclide and magnitude of the body burden

7. identification of the individual making the conclusion.

A sample record sheet is shown in Appendix A.

(4) Airborne-Radioactivity Measurements. If airborne-radioactivity measurements and exposure times indicate that an individual has received an internal exposure via inhalation, the following information should be recorded:

1. identification of the individual

2. period(s) covered by the measurements

3. basis for exposure estimate

4. concentration of airborne radioactive material, length of exposure, and estimated breathing rate

5. reference to any documentation of the factors in item 4

6. estimated internal exposure

7. identification of the investigator.

F. Radiation Exposure Received During Prior Employment. To ensure that the information on DD Forms 1952 and 1141 is complete, the RPO should have each new staff member complete and sign a questionnaire indicating whether any previous employment (civilian or military) may have involved internal or external exposure to radiation, with the names and addresses of former employers where any exposure may have occurred. Previous employers who are contacted for information should be requested to use the individual's social security number when providing information, to ensure the correct identity of the individual.

The following information on each previous exposure should be obtained and kept in the personnel file:

1. the period(s) of employment and the identification of the employer

2. the nature and magnitude of the exposure, both internal and external, and the period of exposure.

G. Exposure Received by Individuals at Other Installations During Current Employment. The radiation exposure received by an individual at another installation during an official visit or special assignment should be maintained in the personnel file. A special film dosimeter may need to be assigned for the visit.

H. Simultaneous Employment at Another Facility. Individuals should report when radiation exposure is being incurred at two facilities as a result of simultaneous employment by two firms or government agencies.

I. Exposure Evaluation. The RPO should review and evaluate DD Form 1141 (or the Automated Dosimetry Records) and the results of any bioassays on a quarterly basis and note the date of the review on DD Form 1141. If action is necessary to limit an individual's exposure, the RPO must notify the individual, the individual's commander and supervisor, and the responsible medical officer.

J. Unusual Exposures. Any accident/incident that involves a radiation worker (such as an exposure in excess of permissible limits, the use of special exposure limits, or an exposure that results in the withdrawal of the individual from a work position--see Chapter 11) must be described and recorded. The extent of the information recorded will depend upon the type of accident/incident but should include:

1. identification of the individual
2. time, date, and location of the accident/incident
3. description of the accident/incident
4. results of the event (e.g., the exposure received by the individual involved, the extent and nature of skin contamination, and any confiscation of personal property)
5. probable cause of the accident/incident
6. action taken at the time of the event
7. reference to or summaries of subsequent action taken to prevent recurrence of the accident/incident

8. reference to or summaries of supporting data used to determine the above items, such as radiation surveys, film dosimeter studies, air sample assays, and photographs

9. identification of the investigator(s).

A sample form is shown in Appendix A.

K. Transfer of Records. When a radiation worker transfers to another assignment or organization, all chargeout records for DD Form 1141 (or the automated dosimetry records) and for bioassay records must be removed from the individual's health record or medical file and replaced with the original forms and records. The health record or medical file, containing complete and accurate originals of DD Form 1952, DD Form 1141 or the automated dosimetry records, and bioassay records, is sent to the gaining organization to which the individual has been assigned. A copy of each document should be retained at the original installation, with the address of the gaining organization noted on the copy of DD Form 1141 to ensure that any additional dosimetry information received after the transfer is forwarded to the gaining organization.

13.1.2 Radiation Protection Program File

A record of the installation's radiation protection policy and procedures should be maintained to allow the RPO and his or her supervisor to continually evaluate and update the program. In addition, records should be readily available to demonstrate to auditors and inspectors the adequacy of the program.

A. Licenses and Authorizations. All documents related to licenses and authorizations to procure and use radioactive materials should be maintained. These documents may include DA permits and authorizations; NRC license applications, licenses, and amendments; and authorizations to store, transfer, ship, or dispose of radioactive materials.

B. Radiation Protection Policies and Standards. Policies and standards established for the overall conduct of radiation work at the installation should be documented. These records should include:

1. scope and organization of the radiation protection program

2. training and experience of the individuals on the radiation protection staff

3. orientation and training requirements for individuals who will perform radiation work

4. specifications for the frequency and techniques to be used in measuring the radiation exposure received by individuals

5. control procedures for radiation work, such as permissible levels of radiation and contamination in work areas, as well as posting and labeling requirements

6. plans and procedures for radiation emergencies, including the type and frequency of training drills

7. criteria for the investigation of unusual radiation occurrences

8. reporting and records requirements

9. regulations, standards, procedures, and higher-headquarters instructions, along with effective dates for each.

C. Documents of the Ionizing Radiation Control Committee. Documents relating to the meetings and decisions of the Ionizing Radiation Control Committee (IRCC) should be kept. This information should include reports on IRCC reviews of applications for approval to use sources of ionizing radiation. The records should note whether each application was approved or disapproved, the conditions under which each source was approved for use, and the qualifications of the users.

D. Procedures for Obtaining and Evaluating Data on Individual Exposures. The procedures used to obtain, process, and evaluate data for individuals' external and internal exposure records should be recorded. Records of the methods used to obtain an individual's exposure should refer to pertinent published documents or reports and should show the period of applicability of the methods used.

E. Inspections and Appraisals. Documents related to compliance inspections performed by DA and civilian licensing agencies should be maintained. These records should include notifications of inspection, inspection reports, and documents related to follow-up corrective actions.

A health physics appraisal provides an evaluation of the overall adequacy and effectiveness of the radiation protection program. Appraisals may be performed by a team of outside experts and/or the installation RPO (see Chapter 15). All of the documents related to the appraisal of the radiation protection program should be maintained and should include appraisal notifications, findings, and corrective actions.

F. Changes in Procedures and Methods. Substantial revisions of procedures, methods of evaluation, or policies should be recorded. When pertinent, the reasons for such changes should also be recorded.

13.1.3 Project File

Each project or task should be fully documented. A title and an identification number should be assigned to a project before it is begun, and project records should be filed by the project identification number.

A. General Records. All documents relating to a project should include the project's title and identification number, key word descriptor(s) relevant to the project, and the name of the principal investigator. A list of key word descriptors available for assignment to a program should also be kept in the project files. The records for each project should include:

1. a complete description of the project with its start and completion dates
2. a complete listing of all radioactive materials used for the project, including for each source its activity, the date the activity was determined, and its half-life
3. a complete listing of all instrumentation used in the project, including for each instrument its identification number (serial or inventory number), company, model number, and storage location
4. the principal investigator, and a list identifying all project workers and the dates on which each individual began and ended work on the project.

Sample project forms are shown in Appendix A.

B. Standing Operating Procedures. Specific procedures performed in connection with a project are described in SOPs. The SOP is a locally developed form completed by the area supervisor and countersigned by the RPO prior to the start of work. The SOP should include:

1. the title and number of the project
2. effective date of the procedure
3. identity of personnel and/or the organization authorized to perform the work
4. location of the work
5. potential radiation hazards and specific procedures, instructions, and precautions to be observed
6. equipment and dosimetry requirements
7. protective-clothing and equipment requirements
8. descriptions of conditions that would terminate or suspend work in progress
9. identity of the individual approving the procedure.

A copy of each SOP initiated for a project should be included in the records for the project and kept in the project file.

13.1.4 Radiation Work Area File

Documentation of work area conditions is necessary to ensure that good housekeeping procedures are followed and that, in the event of an accident/incident, the radiation source could be quickly characterized and doses to personnel in the area estimated with reasonable accuracy.

A. General Records. Any investigation of a radiation accident/incident requires that substantial supportive data be available. The radiation work area file should therefore include for each laboratory or work area:

1. its location and a map showing the layout of the area
2. a description of the uses of the laboratory or area and its facilities (e.g., hoods, glove boxes, permanently installed equipment)
3. the titles and numbers of projects carried out in the area, with the identity of the principal investigator for each.

B. Radiation and Contamination Surveys. Surveys are conducted to assess the condition of a particular work area. Survey records should include:

1. date and time of the survey
2. location of the survey, that is, building and room (sketches may be included)
3. specific location or object surveyed (sketches may be included)
4. purpose of the survey (e.g., leak test of sealed source, routine survey for contamination on floors and other surfaces, or survey to establish dose rates to personnel)
5. identification (type and serial number) of the particular radiation detection instruments used to perform the survey
6. measurement results (e.g., dose rates and contamination levels), and housekeeping conditions observed
7. conclusions and recommendations
8. identification of the individual performing the survey.

C. Area Monitoring Records. Chart recordings of radiation area monitors should identify:

1. period covered by the chart (beginning and ending dates and times)
2. location of the detector and the area monitored
3. a clear relationship between chart divisions and the exposure or exposure rate units
4. identity of the scale or range of operation
5. notations of source checks and calibrations performed
6. identification of the individuals operating the equipment.

Additional information for continuous air monitors should include:

1. type of instrument (e.g., fixed filter or moving tape)
2. tape and chart speed
3. specific relationship between the chart divisions and the concentration of the airborne radioactive material, which depends on the tape speed and flow rate of a moving filter unit, or on the flow rate of a fixed filter unit.

D. Airborne-Radioactivity Monitoring Records. If airborne radioactive material is monitored, the following information should be recorded:

1. date and time of sampling
2. general location of the air-sampling station (building and room)
3. specific location at which the air sample was collected
4. purpose of sample collected (e.g., routine air sampling or air sample for special evaluation)
5. type of sample collection equipment used (e.g, filter, impact, or evacuated ionization chamber)
6. collection efficiency of sampling system
7. flow rate, duration of sampling, and total volume of air sampled
8. identification of sample analysis equipment used
9. counting data: time count was taken, background, source count, gross count, net count, duration of count
10. reference to calculated correction factors such as backscatter, self-absorption, and efficiency of analytical equipment
11. calculated concentration of airborne radioactive material
12. identity of the air contaminant, if determined
13. identification of the individual performing the analysis.

13.1.5 Instrumentation and Dosimeters File

If the limitations of an instrument have not been determined and the instrument has not been calibrated, the information that it provides about radiation levels in work areas is useless. Therefore, records documenting the availability, limitations, and calibration of all radiation-measuring instruments and dosimeters should be kept in an instrumentation and dosimeters file.

A. Capabilities of Dosimeters and Instruments. The following information on the capability of equipment should be recorded:

1. identification, description, and functional specifications of the individually worn dosimeters and the other radiation measurement instruments used in the radiation protection program
2. date and results of any acceptance or performance tests that show the sensitivity, range, and energy dependence of the instruments
3. special studies documenting bases for use, efficiency, correction factors, and interpretation of data.

B. Calibration and Maintenance. Procedures, criteria, and schedules for calibration and maintenance of radiation measurement instruments and dosimeters are of value in demonstrating the instruments' dependability and reliability. Routine survey instruments should be calibrated every 90 days unless subject to extreme environmental conditions, hard usage, or corrosive environments. In these cases, more frequent calibration is required (ANSI N323-1978). Contingency instruments should be calibrated every 240 days. A suspense file can be used for this purpose. The records system should include:

1. procedures used for the calibration of the individually worn dosimeters and other radiation measurement instruments
2. descriptions of the calibration sources and any data showing intercomparisons with sources from other laboratories
3. data on the frequency of calibrations
4. date and results of the calibration tests, including the identification of the individual performing the test
5. maintenance history of individual radiation measurement instruments.

C. Inventory Records. In addition, the following information should be documented for each radiation-measuring instrument and dosimeter:

1. identification: type, company, inventory number
2. manufacturer's specifications
3. titles and numbers of projects for which the instrument has been used
4. person to whom the dosimeter is assigned, and documents used to record issuance and retrieval of dosimeters.

13.1.6 Radioactive-Material Inventory File

The identity, form, activity, and location of each radioactive source must be documented to ensure good housekeeping procedures and provide a quick indication of a lost source. Information on the form and activity of a source can also be used to indicate radiation doses to personnel in the area (in addition to personnel-monitoring devices), particularly for cases where radioactive material was inhaled or ingested.

A. Sealed and Unsealed Sources and Radioactive Commodities. As soon as a radioactive source or commodity is received, a file containing items 1 through 4 below should be set up. Subsequent information that should be kept in this file includes items 5 through 8 below. Items 9 through 11 should be included for radioactive commodities:

1. name of shipper, and DA authorization and NRC license of shipper
2. packing papers that identify the source, the amount and activity of the source, and the date received
3. designated storage location (a subsequent change in storage location, or transfer or disposal of the source, should also be indicated, with the date of the change)
4. department the source is assigned to, and the responsible individual
5. locations and dates of use, identity of involved personnel and (for unsealed sources) quantity used and quantity remaining
6. titles and numbers of projects in which the source or commodity was used
7. leak test records: date, identity of person performing the test, technique used, counting instrument used (with its inventory or serial number), and test results (in dpm, which may be converted to the appropriate curie unit)
8. disposal details - how, when, and where the sources or commodities were disposed of
9. research, development, and test summary

10. associated technical bulletins

11. system safety sources.

A suspense file may be established to schedule leak testing of sealed sources. A sealed-source inventory should list all sealed sources available at the facility, their activity as specified on the packing papers, with the date of receipt, and their storage location. This list should be kept in the inventory file and updated whenever these conditions change, for example, when the storage location is changed, or the source is transferred to another department or disposed of.

Because unsealed sources present both external-contamination hazards and the possibility of internal exposure through inhalation, ingestion, or entry through a wound, it is essential to know how much material is available at any time in a particular location. An unsealed-source inventory should therefore include a list of all unsealed sources available at the site, the quantity and activity of each on the date of its receipt, the storage location of each, and the quantity and activity remaining on the date of any change in a source's location. The total depletion of an unsealed source should be indicated on the inventory.

B. Environmental Samples. Environmental samples (e.g., air, water, soil, vegetation, and game) are often used to characterize the impact of a particular operation on the environment. The samples themselves should be labeled (a numbering system is frequently used) and the records of these samples should include:

1. label identification number
2. type of sample (water, vegetation, etc.)
3. where the sample was obtained
4. counting results
5. instrument used for counting
6. any actions taken as a result of a high reading
7. disposal details - how, when, and where the sample was disposed of.

13.1.7 Waste Management File

Radioactive waste may include sealed or unsealed radioactive sources; contaminated equipment, clothing, and supply items; and biological organs. Chapter 10 provides guidance for the handling, storage, and disposal of low-level radioactive waste. In general, the following items should be documented for radiaoctive waste:

1. assigned identification number(s)
2. physical description of the waste: solid, liquid, or gas, quantity, shipping weight and volume, number of containers, shipping permit number, transport group, package specification and labels used
3. chemical and radioisotope description: hazardous chemicals, solvent present (liquid), radioisotopes present
4. radioactivity and radiation measurements: activity, maximum dose rates at surface and 1 meter, classification
5. identification of previous responsible department or individual and storage location
6. disposal details - how, when, and where the material will be disposed of
7. identification of responsible individual(s).

13.1.8 Transportation File

Any movement of radioactive material onsite or offsite requires careful planning by the shipper and the receiver. Specific documents must accompany the material, and records of all movements must be kept. Shipping procedures, records, and packaging requirements are discussed in Chapter 9. The shipping documents and records described there include:

1. consignee license
2. bill of lading
3. description of material on shipping papers
4. shipper's certification
5. specific instructions for exclusive-use shipments
6. survey records
7. records showing compliance with package design-and-performance standards.

13.1.9 Accidents/Incidents File

Complete records of radiation accidents/incidents are necessary for after-the-fact documentation of the event. The following information about each accident/incident should be recorded:

1. date, time, and location
2. description
3. results of the event (e.g., the exposure received by the individual(s) involved, the extent and nature of skin contamination, and any confiscation of personal property)
4. probable cause
5. action taken at the time of the event
6. reference to or summaries of subsequent action taken to prevent recurrence
7. reference to or summaries of supporting data used to determine the above items, such as radiation surveys, film dosimeter studies, air sample analyses, and photographs
8. identification of the investigator(s).

13.1.10 Training File

The RPO or the training supervisor should maintain a file that includes the following information for each course that is given:

1. date and location of course
2. identity of instructor(s)
3 description of course content, including course outline, syllabus, and other descriptive information
4 identification of individuals in attendance (name, social security number, birth date, sex)
5. results of examinations.

A suspense file can be set up to schedule training or retraining sessions.

13.1.11 Quality Assurance File

Quality assurance programs are described in Chapter 14. Records of each program element should be maintained, including documents related to:

1. facility design
2. procurement
3. organization of the program
4. control of purchased material, equipment, services, and special processes
5. inspections and tests
6. control of measurement and test equipment
7. handling, storage, and shipping procedures for material and equipment
8. nonconformance and corrective actions.

Section 13.2 RECORDS FILING SYSTEM

A recordkeeping system that incorporates the capacity for extensive cross-referencing among files can be invaluable in answering questions and solving problems related to an individual's radiation dose. The 11 files in which the records just described should be kept--personnel file, radiation protection program file, project file, radiation work area file, instrumentation and dosimeters file, radioactive-material inventory file, waste management file, transportation file, accidents/incidents file, training file, and quality assurance file--contain some overlapping information that would permit an individual's work and exposure history to be traced and the conditions under which the individual received any dose to be reconstructed quickly and accurately.

Through a cross-reference system such as that shown in Appendix C, persons who were involved in a project, whether as principal investigator, calibrator of instruments, or radiation surveyor, can be identified and could be called on to assist in the evaluation of exposure trends or the investigation of occurrences. The two flow charts in Appendix C illustrate how this system could be used to solve specific problems. The repetition of some data in more than one

file permits the investigator to track down information by moving from one file to several others, as necessary.

Section 13.3 RECORDS RETENTION AND STORAGE

13.3.1 Types of Records Retention

Records can be kept as hard copy (paper), on a computer disc or tape, or on microfilm or microfiche.[a] The main considerations in choosing which method of retention to use are:

1. the storage space needed for the number of records generated
2. the ease of accessibility to the stored information that each type of record provides
3. the admissibility of each type of record as evidence in a court of law.

The initial expenses of establishing each type of system should also be considered in relation to the long-term gains of the system, but an extensive cost-effectiveness study is beyond the scope of this manual. Each form of record is discussed below in relation to storage needs, accessibility, and legal status.

A. Hard Copy. The American National Standards Institute recommends in its publication ANSI N13.6-1972 that dose records for every individual occupationally exposed to radiation be kept until 10 years after the individual's death (if the date of death is known) or until the individual would have reached the age of 75 (if the date of death is not known). Records should be kept this long for both scientific purposes (to permit studies of the long-term effects of radiation) and legal reasons. An extensive records system for a large program, if kept in hard-copy form, could involve considerable paper and space. Easy access to such a system would require an excellent centralized

---

(a) Microfilm is a fine-grained, high-resolution photographic film containing an image greatly reduced in size from the original. Microfiche is a sheet of microfilm containing multiple microimages in a grid pattern. The term microform is used to refer to any storage form that uses microimages.

filing system. For a small installation with relatively few records, hard-copy files would be practicable. Moreover, in terms of legal applications, hard copy is often the preferred method of records presentation; in a court of law, evidence on original hard copy is difficult to dispute.

B. Computer Records. If computer storage is used, space must be allotted for the computer itself, for a terminal, and for storage of the discs or tapes.

There is a great deal of controversy over the admissibility of a computer printout as evidence in a court of law. It is difficult to guarantee that a program or number has not been tampered with, and the data records cannot be signed as a way of verifying a record or a change in a record. To stand up as legal evidence, computer entries would have to be verified upon entry, and access to the computer would have to be strictly controlled.

C. Microform. Microfilm and microfiche do not take up much space, and a good filing system would allow easy access to records in these forms. Microfilm has the legal status of an original document if it has been made in compliance with the law.(a)

D. Combinations. The use of a combination of record retention systems would provide flexibility and make use of the advantages of each system. A computer system could be used to provide day-to-day access to all types of records, and hard copy or microform could be kept for legal evidence.

The filing system described in this chapter assumes the use of hard copy; however, the concepts discussed could be easily incorporated into a computer or microform file.

13.3.2 Retention Period

The minimum retention period for all the records described in this chapter is 5 years (ANSI 13.6-1972; AR 385-11). However, because records relating to personnel exposure have both scientific and legal implications, the following records for each individual should be kept until the individual would have

---

(a) See the following U.S. Code sections: 44 U.S.C. 3312, 44 U.S.C. 2112.

reached the age of 75 (if the date of death is not known) or until 10 years after his known death (ANSI N13.6-1972):

1. records of internal and external exposures
2. calibration data associated with evaluation of the individual's exposure
3. records of procedures and methods used to interpret and evaluate the individual's exposure
4. records describing unusual occurrences in which the individual was involved.

13.3.3 Storage Precautions

The effort involved in keeping good records would be wasted if they were lost because of fire or theft. To prevent such a loss, the following suggestions are presented:

1. Keep duplicate copies of all vital records in an area remote from the original documents.
2. Use a standard records vault to minimize the possibility of a fire starting in the vault or entering it from outside (National Fire Protection Association 1980).
3. Consider microfilm for records storage after consulting applicable state laws concerning the legal admissibility of microfilm.

REFERENCES

American National Standards Institute (ANSI). 1972. Practice for Occupational Radiation Exposure Records Systems. ANSI N13.6-1966 (R 1972), New York.

American National Standards Institute (ANSI). 1978. Radiation Protection Instrumentation Test and Calibration. ANSI N323, New York.

National Fire Protection Association. 1980. Standard for the Protection of Records. Publication 232, Boston.

U.S. Code. Title 44, Section 2112, "Legal Status of Reproductions; Official Seal."

U.S. Code. Title 44, Section 3312, "Photographs or Micro Photographs Considered as Originals: Certified Reproductions Admissible as Evidence."

U.S. Code of Federal Regulations. 1982. Title 10, Part 19, "Notices, Instruction and Reports to Workers; Inspections." U.S. Government Printing Office, Washington, D.C.

U.S. Code of Federal Regulations. 1982. Title 10, Part 20, "Standards for Protection Against Radiation." U.S. Government Printing Office, Washington, D.C.

U.S. Code of Federal Regulations. 1982. Title 29, Part 570, "Child Labor Regulations, Orders, and Statements of Interpretation." U.S. Government Printing Office, Washington, D.C.

U.S. Code of Federal Regulations. 1982. Title 29, Part 1910, "Occupational Safety and Health Standards." U.S. Government Printing Office, Washington, D.C.

U.S Department of the Army, Headquarters. "Maintenance and Disposition of General Personnel Management and Safety Functional Files." In Army Functional File System. AR 340-18-6, Washington, D.C.

U.S. Department of the Army, Headquarters. Safety - Ionizing Radiation Protection (Licensing, Control, Transportation, Disposal, and Radiation Safety). AR 385-11, Washington, D.C.

U.S. Department of the Army and Defense Logistics Agency. Medical Services - Control and Recording Procedures for Exposure to Ionizing Radiation and Radioactive Materials. AR 40-14, DLAR 1000.28, Washington, D.C.

APPENDIX A

SAMPLE RECORDS FORMS

Personnel File:

- Project Sheet
- Training Record
- Whole-Body-Counter Record
- Radiation Occurrence Record

Project File:

- Project Characterization
- Project Personnel List
- Key Word Descriptors

PROJECT SHEET FOR PERSONNEL FILE

PROJECT SHEET

Name_____ SS#_____ Birth date_____ Sex_____

	Project No.	Project Title	Key Word Descriptor	Date Start	Date End
1.					
2.					
3.					
4.					

TRAINING RECORD FOR PERSONNEL FILE

TRAINING RECORD

Name_____ SS#_____ Birth date_____ Sex_____

	Course Title	Date	Instructor/Location	Test Score
1.				
2.				
3.				
4.				

WHOLE-BODY-COUNTER RECORD FOR PERSONNEL FILE

WHOLE-BODY-COUNTER RESULTS

Name_____ SS#_____ Birth date_____ Sex_____

Date of Measurement_____

Purpose
_____ routine
_____ suspected intake: see below*

Type of Measurement (check):
_____ whole body
_____ lung
_____ thyroid

Radionuclide (check)	Count Rate (cpm)	Activity (dpm or μCi)	Body Burden (dpm or μCi)
_____ ^{241}Am			
_____ ^{226}Ra			
_____ ^{222}Rn			
_____ ^{235}U			
_____ ^{234}Th			

Instrument used (name, company, model number, identification number): _____

Calculational Method: cpm to dpm or μCi

* Date of suspected intake: _____

Location of suspected intake: _____

Project number, title, principal investigator:_____

RADIATION OCCURRENCE RECORD FOR PERSONNEL FILE

RADIATION OCCURRENCE REPORT

Name_____ SS#_____ Birth date_____ Sex_____

Occurrence Time and Date _____ Building and Location _____

Occurrence Reported by _____

Air Sample ID Number _____ Dosimeter ID Number _____ Survey ID Number _____

Other supporting data (description and location)

Occurrence Description:

Probable Cause:

Initial Actions:

Subsequent Actions to Prevent Recurrence:

Radiation Exposure Data
(check) _____ α _____ β _____ γ _____ n Dose _____(rem)

(check) _____ Skin contamination _____ Internal deposition _____ First aid
_____Hospitalization
Describe:

Investigated by_____

Date_____

PROJECT CHARACTERIZATION FOR PROJECT FILE

PROJECT CHARACTERIZATION

Project No._____ Key Word Descriptor _____

Title _____ Principal Investigator _____

Start Date_____ Ending Date_____

Location of Work:

Description:

Instrumentation (I.D. Number, company, model number, storage location)

Radioactive Materials

Identity	ID No.	Sealed or Unsealed	Location	Radiation(s)	Activity and Date	Half-life

---

HP Support

Dosimetry required_____

Monitoring required_____

Protective equipment required_____

Special instructions_____

_____

_____

PROJECT PERSONNEL LIST FOR PROJECT FILE

PROJECT PERSONNEL RECORD

Project No._____ Key word descriptor_____

Title_____ Principal investigator_____

Name of Radiation Worker	Social Security Number	Date (Start)	Date (End)

KEY WORD DESCRIPTORS FOR PROJECT FILE

KEY WORD DESCRIPTOR LIST

Key Word Descriptor	Project Number and Title	

APPENDIX B

OCCUPATIONAL RADIATION EXPOSURE FORMS

Department of Defense Form 1952

Department of Defense Form 1141

DOSIMETER APPLICATION AND RECORD OF OCCUPATIONAL RADIATION EXPOSURE

Print legibly or type all information requested. See Privacy Act Statement on reverse.

1. FULL NAME (Last, First, Middle)	2. DATE OF BIRTH (YYMMDD)	3. SOCIAL SECURITY NO.
JARVIS, Whitney N.	42-04-15	777-07-3000

4. DUTY SECTION (Dept., Ward, Unit, etc.)	5. JOB TITLE	6. DUTY PHONE
Research Laboratory	Chemist	283-1814

7. PAY GRADE		8. HAVE YOU WORN A DOSIMETER ISSUED BY THIS COMMAND IN THE PAST	9. DATE OF RADIATION PHYSICAL (YYMMDD)
CIVILIAN	MILITARY		
GS-12		☐ YES ☒ NO	81-05-01

10. DUTY STATUS	IF TRANSIENT SHOW MAILING ADDRESS (street address, city, state, zip code) OF LOCATION OF HEALTH RECORDS
☒ PERMANENT ☐ TRANSIENT 8 WEEKS OR LESS	

EXPOSURE INFORMATION *(ITEMS 11 THROUGH 20 FOR HEALTH PHYSICS USE ONLY)*

11. CLASSIFICATION OF EXPOSURE: ☒ EXTERNAL ☐ NEUTRON ☐ INTERNAL

12. BADGES REQUIRED: ☐ WRIST ☒ WHOLE-BODY ☐ NEUTRON

13. TLD REQUIRED: ☐ WRIST ☐ WHOLE-BODY ☐ FINGER

14. BIOASSAYS REQUIRED

WHOLE-BODY COUNT	THYROID UPTAKE	URINALYSIS	FREQUENCY
☐ YES ☐ NO	☐ YES ☐ NO	☐ α ☐ β ☐ β-γ	☐ MONTHLY ☐ QUARTERLY ☐ ANNUALLY

GIVE DATES FOR ITEMS 15 THROUGH 20 (YYMMDD)

15. DOSIMETER(S) ISSUED	16. DD FORM(S) 1141 INITIATED	17. DOSIMETER(S) DISCONTINUED
81-05-03	81-05-03	
18. LAST DOSIMETER(S) RETURNED	**19. LOCATOR CARD TO HEALTH RECORD**	**20. DD FORM(S) 1141 TO MEDICAL RECORDS**
	81-05-03	

OCCUPATIONAL EXPOSURE HISTORY

NOTE: This section only applies to the individual who has worked with radiation-producing devices or radioisotopes in a permanent status. List only those employers for whom you worked with radiation.

NAME OF EMPLOYER	ADDRESS (street address, city, state, zip code)	FROM YR	FROM MO	TO YR	TO MO	Do not write in this space
Nuclear Services, Inc	Shickshinny, PA	78	08	80	04	
Rosewater University	Portland, OR	80	04	81	04	
	TOTAL EXPOSURE DATA					

REMARKS

DD FORM 81 NOV 1952 EDITION OF 1 SEP 74 IS OBSOLETE. (see reverse)

PRIVACY ACT STATEMENT
DATA REQUIRED BY THE PRIVACY ACT OF 1974
(5 USC 552a)

1. TITLE OF FORM: Dosimeter Application and Record of Occupational Radiation Exposure.

2. PRESCRIBING DIRECTIVE: AR 40-14 and DLAR 4145.24.

3. AUTHORITY: 5 USC 301-Departmental Regulation; 10 USC 1071, Medical and Dental Care, Purposes; 42 USC 2073, 2093, 2095, 2111, 2133, 2134, 2201(b), and 2201(o). The authority for soliciting the social security number is 10 CFR 20; 44 USC 3101-Record Management by Agency Heads, General Duties.

4. PRINCIPAL PURPOSE(S): To establish qualification of personnel monitoring and document previous exposure history. The information is used in the evaluation of risk of exposure to ionizing radiation or radioactive materials. The data permits meaningful comparison of both current (short-term) and long-term exposure to ionizing radiation or radioactive material. Data on your exposure to ionizing radiation or radioactive materials is available to you upon request.

5. ROUTINE USES: The information may be used to provide data to other Federal agencies, academic institutions, and non-governmental agencies, such as the National Council on Radiation Protection and Measurement and the National Research Council, involved in monitoring/evaluating exposures of individuals to ionizing radiation or radioactive materials who are employed as radiation workers on a permanent or temporary basis and exposure received by monitored visitors. The information may also be disclosed to appropriate authorities in the event the information indicates a violation or potential violation of law and in the course of an administrative or judicial proceeding.

6. MANDATORY OR VOLUNTARY DISCLOSURE AND EFFECT ON INDIVIDUAL NOT PROVIDING INFORMATION: It is voluntary that you furnish the requested information, including social security number; however, the installation or activity must maintain a completed DD Form 1141 on each individual occupationally exposed to ionizing radiation or radioactive material as required by 10 CFR 20, 29 CFR 1910.96 and AR 40-14/DLAR 4145.24. If information is not furnished, individual may not become a radiation worker. The social security number is used to assure that the Army/Agency has accurate identifier not subject to the coincidence of similar names or birthdates among the large number of persons on whom exposure data is maintained.

STATEMENT

Under the provisions of 10 CFR 19.13, 29 CFR 1910.96 and the Privacy Act of 1974, I hereby authorize the release of, and request that all of my radiation exposure records be furnished appropriate authorities in accordance with the "Routine Uses" portion of the above Privacy Act Statement. As a radiation worker, I have been provided instructions in radiation protection as required by 10 CFR 19.12 and 29 CFR 1910.96. As a female radiation worker, I have been informed of the biological affects and the risks from ionizing radiation on the embro-fetus and received a copy of NRC (Nuclear Regulatory Commission) Guide 8.13. I will contact my supervisor or the radiation protection officer if I have any questions. I hereby certify that the exposure history listed on the obverse is correct and complete, to the best of my knowledge and belief. I have read and understand the above Privacy Act Statement.

81-04-25
Date (YYMMDD)

Whitney M. Jarvis
Signature of Applicant

RECORD OF OCCUPATIONAL EXPOSURE TO IONIZING RADIATION

FOR INSTRUCTIONS, SEE REVERSE OF SHEET.

1. IDENTIFICATION NUMBER	2. NAME (Last, first, middle initial)	3. SOCIAL SECURITY NUMBER	4. RANK/RATE TITLE OF POSITION	5. DATE OF BIRTH (Day, month, year)
074	JARVIS, WHITNEY N.	777-07-3000	TDR	15 Apr 42

PLACE WHERE EXPOSURE OCCURRED	PERIOD OF EXPOSURE		DOSE THIS PERIOD (rem) 1. Method of monitoring is presumed to be film badge reading unless otherwise specified under Item 16, "REMARKS."				ACCUMULATED DOSE (rem)		INITIAL
ACTIVITY	FROM (Day-Mo-Yr)	TO (Day-Mo-Yr)	SKIN DOSE (Salt)	GAMMA AND X-RAY	NEUTRON	TOTAL THIS PERIOD	TOTAL LIFETIME	PERMISSIBLE LIFETIME 5(N-18)	PERSON MAKING ENTRY
6	7	8	9	10	11	12	13	14	15
WHOLE BODY									
Previous Exposure[1]	Aug66	Apr68	NR	00.107	NU	00.107	00.107	-	CED
Admin Dose[3]	Apr68	Apr69	-	-	-	05.000	05.107	45.000	CED
APG-EA, MD	3May69	4Jun69	NR	00.000	NU	00.000	05.107	45.000	CED
do	6Jun69	6Jun69	Quarterly Review by RPO				-	-	JER
do	5Jun69	4Jul69	00.003	00.010	NU	00.010	05.117	45.000	CED
do	5Jul69	7Aug69	NR	00.078	NU	00.078	05.195	45.000	CED
do	8Aug69	6Sep69	Film Badge Lost[3]		NU	00.416	05.611	45.000	CED
do	8Sep69	8Sep69	Quarterly Review by RPO				-	-	JER
do	7Sep69	4Oct69	NR	00.064	NU	00.064	05.675	45.000	CED
do	5Oct69	4Nov69	NR	00.075	NU	00.075	05.750	45.000	WLW
do	5Nov69	6Dec69	00.016	00.070	NU	00.070	05.820	45.000	WLW
do	Film Badge Service Discontinued 6 Dec 69						-	-	WLW
do	6Dec69	6Dec69	Quarterly Review by RPO				-	-	JER
Fort Plunkett	2Jan70	3Feb70	NR	00.000	00.000	00.000	05.820	45.000	RKO
do	4Feb70	3Mar70	NR	00.178	00.062	00.240	06.060	45.000	RKO
do	4Mar70	2Apr70	00.052	02.504	00.126	02.630	08.690	45.000	RKO
do	22Mar70	22Mar70	Quarterly Review by RPO				-	-	MJM
do	3Apr70	4May70	Relieved From Duties				08.690	50.000	RKO
do	5May70	3Jun70	Involving Exposure to RAD[5]				08.690	50.000	RKO
do	4Jun70	2Jul70	00.017	00.100	00.043	00.143	08.833	50.000	RKO
Fort Smith, CA	Aug70	Jul71	No Film Badge Worn or Exposure Received				08.833	55.000	GML
			SAMPLE						

16. REMARKS (Continue on additional sheet if necessary)

1. Nuclear Services, Inc., Shickshinny, PA
2. Rosewater University, Portland, OR
   No film badge records (AR 40-14).
3. Admin Dose = $\frac{5 \text{ rem}}{12 \text{ months}}$ = 00.416 rem
4. Alleged overexposure.
5. Pending investigation IAW AR 40-5.

NR - none reported; NU - not used
Has wrist badge No. 086.

TO BE RETAINED PERMANENTLY IN INDIVIDUAL'S MEDICAL RECORD

DD FORM 1 MAY 67 1141 PREVIOUS EDITIONS ARE OBSOLETE.

RECORD OF OCCUPATIONAL EXPOSURE TO IONIZING RADIATION

FOR INSTRUCTIONS, SEE REVERSE OF SHEET.

1. IDENTIFICATION NUMBER	2. NAME (Last, first, middle initial)	3. SOCIAL SECURITY NUMBER	4. RANK/RATE TITLE OF POSITION	5. DATE OF BIRTH (Day, month, year)
086	JARVIS, WHITNEY N.	777-07-3000	TDR	15 Apr 42

PLACE WHERE EXPOSURE OCCURRED	PERIOD OF EXPOSURE		DOSE THIS PERIOD (rem) 1. Method of monitoring is presumed to be film badge reading unless otherwise specified under Item 16, "REMARKS."				ACCUMULATED DOSE (rem)		INITIAL
ACTIVITY	FROM (Day-Mo-Yr)	TO (Day-Mo-Yr)	SKIN DOSE (Self)	GAMMA AND X-RAY	NEUTRON	TOTAL THIS PERIOD	TOTAL LIFETIME	PERMISSIBLE LIFETIME 5(N-18)	PERSON MAKING ENTRY
6	7	8	9	10	11	12	13	14	15
Previous Exposure[2]	Aug66	Apr68	-	-	-	00.204	00.204	NA	CED
Admin Dose[3]	Apr68	Apr69	-	-	-	75.000	75.204	NA	CED
APG-EA, MD	3May69	4Jun69	NR	00.009	NU	00.009	75.213	NA	CED
do	6Jun69	6Jun69	Quarterly Review by RPO				-	NA	JER
do	5Jun69	4Jul69	00.007	00.018	NU	00.018	75.231	NA	CED
do	5Jul69	7Aug69	NR	00.159	NU	00.159	75.390	NA	CED
do	8Aug69	6Sep69	Film Badge Lost[4]		NU	06.250	81.640	NA	CED
do	8Sep69	8Sep69	Quarterly Review by RPO				-	NA	JER
do	7Sep69	4Oct69	NR	00.143	NU	00.143	81.783	NA	CED
do	5Oct69	4Nov69	NR	00.162	NU	00.162	81.945	NA	WLW
do	5Nov69	6Dec69	00.032	00.150	NU	00.150	82.095	NA	WLW
do	Film Badge Service Discontinued 6 Dec 69						-	NA	WLW
do	6Dec69	6Dec69	Quarterly Review by RPO				-	NA	JER
Fort Plunkett	2Jan70	3Feb70	NR	00.015	NU	00.015	82.110	NA[F]	RKO
do	4Feb70	3Mar70	NR	00.420	NU	00.420	82.530	NA	RKO
do	4Mar70	2Apr70	00.140	18.125[5]	NU	18.125	100.655	NA	RKO
do	22Mar70	22Mar70	Quarterly Review by RPO				-	NA	MJM
do	3Apr70	4May70	Relieved From Duties[6]				100.655	NA	RKO
do	5May70	3Jun70	Involving Exposure to RAD				100.655	NA	RKO
do	4Jun70	2Jul70	00.025	00.200	NU	00.200	100.855	NA	RKO
Fort Smith, CA	Aug70	Jul71	No Film Badge Worn or Exposure Received				100.855	NA	GML
			S A M P L E						

16. REMARKS (Continue on additional sheet if necessary)

1. Wrist Record (WB Record 074)
2. Nuclear Services, Inc., Shickshinny, PA
3. Rosewater University, Portland, OR
   No film badge records (AR 40-14)
   NR - none reported; NU - not used.
4. Admin Dose = $\frac{75 \text{ rem}}{12 \text{ months}}$ = 06.250
5. Accidental Exposure. Case documented IAW AR 40-5.
6. Necessary to avoid exceeding quarterly limit

TO BE RETAINED PERMANENTLY IN INDIVIDUAL'S MEDICAL RECORD

DD FORM 1 MAY 67 1141 PREVIOUS EDITIONS ARE OBSOLETE.

APPENDIX C

CROSS-REFERENCE SYSTEM FOR FILES, AND FLOW CHARTS FOR PROBLEM SOLVING

TABLE 13.1. Cross-Reference System

Records	Cross-Reference
Personnel File	
Identification of the Radiation Worker	
Radiation Exposure Received During Prior Employment	
Exposure Received by Individuals at Other Installations During Current Employment	
Simultaneous Employment at Another Facility	
Training Records	Training File
Project/Task Listing	Project File
External-Exposure Records	Instrumentation and Dosimeters File
Internal-Exposure Records	Instrumentation and Dosimeters File
Exposure Evaluation	Radiation Protection Program File
Unusual Exposures	Accident/Incident File
Radiation Protection Program File	
Licenses and Authorizations	
Radiation Protection Policies and Standards	
Procedures and Methods for Interpretation and Evaluation of Individual Exposure Data	
Inspections and Appraisals	
Changes in Procedures and Methods	

TABLE 13.1. (continued)

Records	Cross-Reference
Project File	
General Records	
- project descripton--dates, location	Radiation Work Area File
- radioactive materials list	Radioactive-Material Inventory File
- instrumentation list	Instrumentation and Dosimeters File
- principal investigator/project workers	Personnel File
Standing Operating Procedures	
Radiation Work Area File	
General Records	
- location/map	
- work area uses/equipment and instruments	Instrumentation and Dosimeters File
- projects in the area	Project File
- project principal investigator	Personnel File
Radiation and Contamination Surveys	
- date, time, location, purpose	
- instrument identification	Instrumentation and Dosimeters File
- measurement results	Radiation Protection Program File
- individual(s) performing survey	Personnel File
Area Monitoring Records	
- date, location	
- instrument type, calibration	Instrumentation and Dosimeters File
- source check records	Radioactive Material Inventory File
- individual(s) operating equipment	Personnel File
Airborne-Radiation Monitoring Records	
- date, time, location, purpose	
- identity of sampling equipment	Instrumentation and Dosimeters File
- collection efficiency	Instrumentation and Dosimeters File
- counting data	
- calculated correction factors, concentrations, and efficiency of equipment	Radiation Protection Program File

TABLE 13.1. (continued)

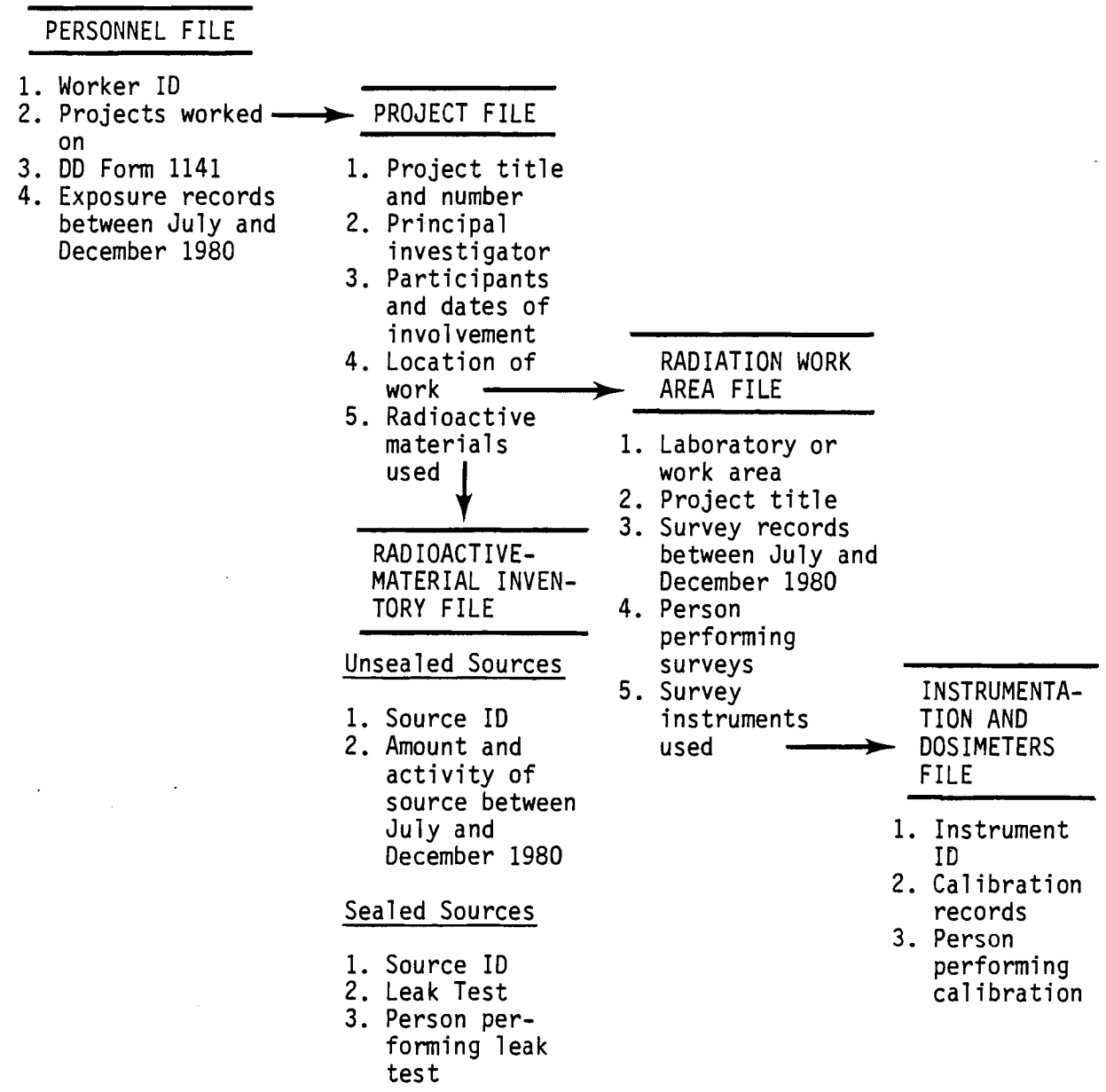
Records	Cross-Reference
Radiation Work Area File (continued)	
- identity of air contaminant	
- individual performing analysis	Personnel File
Instrumentation and Dosimeters File	
Capabilities of Dosimeters and Instruments	
Calibration and Maintenance	Personnel File Radioactive-Material Inventory File
Inventory Records	Project File
Radioactive-Material Inventory File	
Sealed Sources	
- packing papers	
- storage and use locations	Radiation Work Area File
- responsible department/individual	Personnel File
- projects	Project File
- project personnel	Personnel File
- leak test records:	
- instrument	Instrumentation and Dosimeters File
- individual	Personnel File
- results	Radiation Protection Program File
- disposal history	Waste Management File
- inventory	
Unsealed Sources	
- packing papers	
- storage and use locations	Radiation Work Area File
- responsible department/individual	Personnel File
- dates of use, quantity	
- projects	Project File
- project personnel	Personnel File
- disposal history	Waste Management File
- inventory	
Environmental Samples	
- identification number, sample type	
- location	
- counting results	Radiation Protection Program File

TABLE 13.1. (continued)

Records	Cross-Reference
Radioactive Material Inventory File (continued)	
- counting instrument	Instrumentation and Dosimeters File
- disposal history	Waste Management File
Waste Management File	
General Records	
- assigned identification number	Radioactive-Material Inventory File
- physical description	
- chemical and radioisotope description	
- radioactivity and radiation measurements	
- previously responsible department/individual(s)	Personnel File
- storage location	Radiation Work Area File
- disposal details:	
- how, when, where	
- responsible individual	Personnel File
Transportation File	
Radioactive-Material Shipments	Radioactive-Material Inventory File
Accidents/Incidents File	
General Records	
- date and time	
- location	Radiation Work Area File
- description, cause	
- involved individual(s)	Personnel File
- corrective/protective actions	
- supporting data:	
- survey, sample results	Radiation Work Area File
- instruments, dosimeters	Instrumentation and Dosimeters File
- investigator(s)	Personnel File
Training File	
General Records	
- date	
- instructor/attendees	Personnel File
- description	

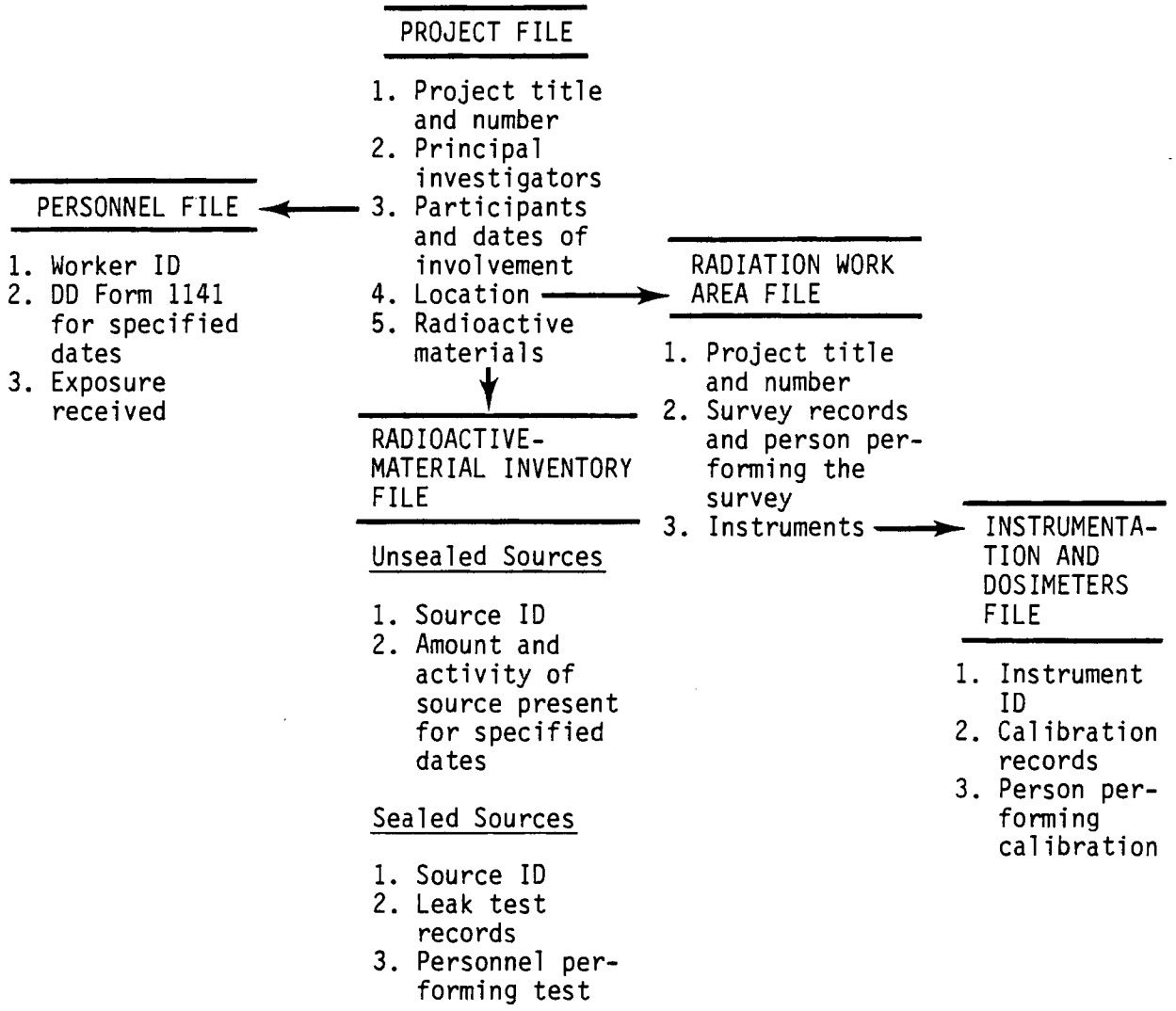
FLOW CHART 1. Occupational Exposure History

Problem: Recreate staff member's working conditions and verify exposure from July through December 1980.



FLOW CHART 2. Project Characterization

Problem: Confirm or refute allegations of misuse of radioactive materials during a specific project that could have resulted in overexposures.



CHAPTER 14. QUALITY ASSURANCE PROGRAM

CHAPTER 14. QUALITY ASSURANCE PROGRAM

The purpose of a radiation protection program is to provide control in the storage, handling, and use of radioactive material and radiation-generating machines, so as to minimize the hazard to personnel and the general public. Personnel responsible for the radiation protection program must implement established regulations and meet the requirements of the facility license. They are also responsible for ensuring that the radiation protection program accomplishes its purpose. Consequently, a surveillance plan is needed to verify that activities are conducted as desired and that regulations are met. A quality assurance program provides a means of controlling the radiation protection program and verifying that it is meeting the purposes for which it was established. It allows those responsible for a program or a facility to ensure that the quality required for safe and reliable operation is achieved.

This chapter provides a review of the elements of quality assurance and how they are incorporated into a radiation protection program. Special terms are defined near the beginning of the chapter, followed by a discussion of how a quality assurance program is implemented--when a program is needed and how extensive it should be. The elements of a quality assurance program, including the purpose of each element and the activities it involves, are then reviewed.

Section 14.1 QUALITY ASSURANCE AND QUALITY CONTROL

Quality assurance is sometimes confused with quality control. Quality assurance is all of the planned and systematic actions needed to provide adequate confidence that a structure, system, or component will perform satisfactorily in service. In other words, quality assurance is a planned program for verifying that each part of the radiation protection program is being carried out adequately and that the total program meets its purpose. It is the application of systematic management principles, such as planning, documenting, auditing, and verifying. Quality control is the quality assurance actions that relate specifically to the physical measurement of an

item, and it provides a means of controlling the quality of the item to predetermined requirements. Quality control is a part of quality assurance.

While quality performance is the responsibility of each individual, a planned quality assurance program provides a method of 1) ensuring that all the elements necessary for adequate radiation protection have been considered, and 2) verifying their implementation. The installation commander designates who is responsible for the quality assurance program, for example, a Quality Assurance Office, Plans and Programs Office, or Program Evaluation Office.

Section 14.2 DEFINITIONS

Some terms have a specific meaning when used in quality assurance programs. The terms used in this chapter are defined below.

1. Quality assurance - All of the planned and systematic actions needed to provide adequate confidence that a structure, system, or component will perform satisfactorily in service.
2. Quality control - The quality assurance actions that control the physical measurements of an item in accordance with predetermined requirements.
3. Analysis - The examination of a complex problem by separating it into its fundamental elements.
4. Appraisal - The evaluation of the worth, significance, or status of a program or item.
5. Audit - A formal, documented examination of an activity or program to verify compliance with established requirements.
6. Evaluation - The determination of the worth of something by careful appraisal and study.
7. Inspection - Examination or measurement to verify whether an item or activity conforms to specified requirements.
8. Surveillance - Monitoring or observation to verify whether an item or activity conforms to specified requirements.

9. Test - The determination of the capability of an item to meet specified requirements by subjecting the item to a set of physical, chemical, environmental, or operating conditions.

Section 14.3 IMPLEMENTATION OF QUALITY ASSURANCE

Because different operations involve different degrees of risk, not all facilities or operations require the same degree of quality assurance application. The level of control and assurance necessary for a specific facility or operation depends upon the importance and complexity of the operation and its effect on the safety of the facility, its personnel, and the public. For example, the quality assurance program for an instrument calibration facility requires rigorous control and documentation to ensure that instrument measurements are accurate and reproducible; control of radiation sources to ensure that they are traceable to nationally recognized standards; and records to ensure that regulations on the quality of instrument calibration and frequency are met. In contrast, an operation involving the use of a commercial device with an internal, sealed radioactive source may require only a periodic inventory to verify the location of the device, and a routine wipe survey to ensure that the source is intact and not leaking.

14.3.1 Who Needs a Quality Assurance Program

A quality assurance program should be developed for facilities or locations where the following take place:

1. radioactive material is received, stored, handled, or used
2. radiation-generating machines are operated
3. personnel radiation dosimetry is evaluated
4. radiation detection or measurement equipment is procured, received, repaired, calibrated, or used
5. facilities or equipment that will be used for these operations are designed, constructed, or modified.

Each facility or operation should have, as a minimum, a written quality assurance program that defines the extent and content of the program, the records required, and the audit activities needed to verify implementation of the program.

14.3.2 How Extensive a Program Should Be

Not every quality assurance program requires additional staff or a rigorous effort. The extent of the quality assurance program needed for a facility or operation should be determined from a thorough evaluation of the activities to be conducted, their potential effect on the safety of plant personnel and the public, and the requirements of applicable regulations and licenses. The quality assurance program should provide documented, verifiable evidence to support the reliability and effectiveness of the radiation safety program, and compliance with regulatory and license requirements.

Radiation protection actions for which a quality assurance program should be developed include, but are not limited to:

1. dose evaluation for all personnel who work at the facility and for all visitors
2. receipt, inventory, shipping, and disposal of radioactive material
3. radiation and contamination surveys
4. detection, measurement, and evaluation of airborne radioactivity
5. procurement, receipt, maintenance, repair, and calibration of radiation detection and measurement equipment
6. personnel qualification, training, and retraining
7. radioactive-effluent releases and environmental monitoring
8. facility design and modification
9. abnormal occurrences and investigations of them.

14.3.3 Who Determines the Extent of the Program

The extent of a quality assurance program should be determined by the manager responsible for the overall performance and safety of a facility or operation. In most instances, this responsibility is assigned one level above

the person responsible for the facility's radiation protection program. The person responsible should use the general guidelines provided above to develop a quality assurance program as extensive as is needed to assure adequate radiation protection. In some instances, parts of the quality assurance program may be established by regulatory requirements or by the recommendation of the Ionizing Radiation Control Committee.

Section 14.4 ELEMENTS OF A QUALITY ASSURANCE PROGRAM

A quality assurance program is composed of numerous elements. Each one is intended to provide surveillance of a major aspect of the radiation protection program. All elements may not be needed in a particular facility or operation, but each should be considered when the quality assurance program for that facility is being established. How far each element of the quality assurance program is developed depends upon the degree of control required.

14.4.1 Organization of the Quality Assurance Program

A defined organizational structure should be established for the quality assurance program to ensure the effective management of quality assurance activities. The organizational structure, functional responsibilities, levels of authority, and lines of communication for operations affecting the quality of the radiation protection program should be written down. All individuals in the program should know what their jobs are, what authority they have to accomplish their work, and to whom they should report problems so that corrective action will be taken.

The person or organization responsible for developing and implementing the quality assurance program should be specified. This person or organization should have sufficient authority, access to work areas, and organizational freedom to 1) identify quality problems; 2) recommend or provide solutions to quality problems through designated channels; 3) verify that the solutions have been implemented; and 4) ensure that any further processing, delivery, installation, or operation is controlled until the problem has been corrected. The person or organization should have direct access to responsible management at a level where appropriate action can be taken.

The organizational structure should be designed and individual responsibilities should be assigned so that quality is achieved and maintained by those responsible for a job and is verified by persons or organizations not directly responsible for the job. In some instances, it may not be possible or practical to assign a separate organization or person to verify the achievement of quality. In those instances, the quality assurance responsibilities should be written carefully to ensure that they do not conflict with the job responsibilities of the individual assigned to carry out multiple duties. It may be appropriate to have an outside organization provide quality assurance. In all cases, the individual's or organization's responsibility and authority should be clearly defined and documented.

14.4.2 Preparation and Documentation of the Quality Assurance Program

The quality assurance program should be documented as a means of defining the program, providing a basis for review, and ensuring continuity. Adequate planning is needed before the quality assurance document is written to ensure that all necessary elements have been included. The program should:

1. provide control over operations affecting the quality of the radiation protection program, to whatever extent is consistent with the importance of those operations
2. identify the operations, processes, and equipment to which the program applies
3. include consideration of the technical aspects of quality assurance actions
4. be established as early as possible consistent with the schedule for accomplishing the operation
5. provide for any special controls, processes, test equipment, tools, and skills needed to attain the required quality and for necessary verification of quality
6. provide for the training of personnel performing operations that affect quality, to ensure that they can do the job adequately

7. provide for regular management assessment of the adequacy and effective implementation of the quality assurance program.

Not all operations in a facility or program require formal quality assurance consideration. Those operations that are important for adequate radiation protection and/or that must be performed consistently should be included in the quality assurance program plan. The program should specify the qualifications, training, and skills required of quality assurance personnel; the type, frequency, and method of audits, inspections, and tests for the assurance of quality; and the system for reporting, correction, and follow-up on any unsatisfactory condition that may be identified. If an extensive quality assurance program is necessary, additional details on planning the program may be found in the American National Standards Institute's (ANSI's) standards and in the Nuclear Regulatory Commission's (NRC's) regulatory guides. (See the bibliography at the end of this manual.)

14.4.3 Control of Facility Design

It is particularly important that radiation protection requirements be included in the design of new facilities or the modification of existing facilities in which radioactive material will be stored, handled, or processed or in which radiation-generating machines will be operated. Engineered features for controlling radiation and contamination are most cost-effective, and some are only feasible, when included in the original design and construction or in a major modification of a facility. The design for facilities should therefore be defined, controlled, and verified to ensure that radiation protection requirements are met, that the design is approved by appropriate authorities, and that construction meets the design specifications.

A. Designs for Facilities. Appropriate design bases, performance requirements, regulatory requirements, and codes and standards should be identified and documented, and their selection reviewed and approved. For radiation protection purposes, the ventilation criteria, shielding provisions, equipment reliability and maintenance, personnel traffic patterns and occupancy zones, and waste-handling systems must be reviewed specifically for how well they will protect personnel and keep radiation doses as low as is reasonably achievable (ALARA). If designs are changed, the changes and the reason for the

changes should be identified, approved, and documented to ensure that alterations that could affect radiation protection are adequately reviewed. The purpose of quality assurance in this process is to verify that these steps are taken and that reviews and approvals are completed by appropriate personnel.

The organization responsible for the design should document its actions in enough detail so that the design process can be carried out and it is possible to verify that the design meets requirements. The design of the facility, and the materials, equipment, and processes that are essential to radiation protection and exposure control, should be selected and reviewed for suitability of application.

B. Independent Analysis of Designs. It may be advisable to provide for an independent analysis of the design of facilities and equipment, to ensure that all factors have been considered and that the resulting designs are correct. The independent analysis should be performed by individuals who are technically qualified in the subject and independent of the original designers. These persons may vary from electrical experts, who ensure that electrical load-carrying capacities are adequate, to health physicists, who ensure that shielding factors for shielding casks are correct.

The analysis should be documented in enough detail so that a person technically qualified in the subject can review the analysis and verify the findings. Documentation of an analysis should include the purpose of the analysis, pertinent sources of data and supporting information, and review and approval. Here again, the quality assurance function is to verify that analyses, reviews, and approvals required as part of the quality assurance program have been completed.

C. Design Verification. Designs for important facilities should be verified to ensure that the design was performed correctly and that the final product as provided for in the design will perform the function described in the design criteria.

Designs should be verified by competent personnel other than those who drew up the original design. The design verification results should be documented and the verifier identified. The extent of the design verification

required depends on the importance to safety of the item under consideration, and verification methods may include analyses, simple reviews, alternate calculations, and/or qualification testing.

D. Design Changes and Documentation. Once a design has been approved, any changes, including field changes, should be controlled in the same manner as the original design.

The design documentation and records, which provide evidence that the facility was designed and the design was verified as required, should be generated and maintained in accordance with documented procedures.

14.4.4 Control of Procurement Documents

A fourth function of a quality assurance program is to ensure that documents generated for the procurement of items or services include enough information (applicable design bases, technical requirements, specifications, drawings, instructions, etc.) so that the items being procured will be adequate in quality; they must fit, work properly, and do the job required.

The procurement documents should identify the means (tests, inspections, documentation) that the purchaser will use to determine the acceptability of the items. If certain aspects of acceptability cannot be determined at this point, the procurement documents should specify the quality assurance requirements necessary in the supplier's plant.

Procurement documents and changes to them should be reviewed and approved by the purchaser to ensure that they are clear and detailed enough so that the supplier can provide the items or services that meet the specified requirements.

Depending upon the type and use of the item or service being procured, it may be necessary to include in procurement documents the requirement that suppliers also have and implement a documented quality assurance program.

14.4.5 Instructions, Procedures, and Drawings

Operations that affect the quality of the radiation protection program must be reproducible, and complex operations should be performed in accordance with documented instructions, procedures, or drawings as appropriate to the circumstances, to ensure consistent and adequate performance. Such operations include tests, equipment control, calibration of instruments, and surveys.

14.4.6 Document Control

Documents that specify quality requirements or prescribe operations affecting quality should be prepared, issued, and changed in a controlled manner to ensure that correct documents are being used. These documents, including changes to them, should be reviewed for adequacy and approved for release by authorized personnel.

The document control system should identify which documents are to be controlled; who is responsible for preparing, reviewing, approving, and issuing them; how their adequacy, completeness, and correctness is to be ensured prior to issuance; and the methods of ensuring that documents in use are current and that outdated or inappropriate documents are removed from use.

14.4.7 Control of Purchased Material, Equipment, and Services

The procurement of material, equipment, and services should be controlled to ensure conformance with the requirements specified in the procurement documents. Procurement operations should be planned and documented and should include the preparation and review of procurement documents and control of changes to them (see item 14.4.4 above); selection of procurement sources; the evaluation of bids and the award of a contract; purchaser control of supplier performance, if warranted by the circumstances; any necessary verification actions, including surveillance, inspection, or audit of the supplier; plans for controlling and disposing of material, equipment, or services that do not meet requirements; methods of correcting problems occurring in the procurement process; acceptance of material, equipment, or services; and the quality assurance records needed. Most purchased material, equipment, and services should be inspected when they are received from the supplier to ensure that they meet the requirements of the procurement documents and the purpose for which they were purchased.

14.4.8 Material Identification Control

Controls should be established to ensure that only correct and accepted items are used or installed. Identification should be maintained either on the items or in documents traceable to the items.

Items with a limited calendar or operating life should be identified and controlled to prevent their use after their life has expired. For instance, batteries, some adhesives, rubber products, chemicals, and radioactive sources may degrade in storage as well as in use and may need to be controlled to ensure their effectiveness when needed.

14.4.9 Control of Special Processes

Measures should be established and documented to ensure that special processes such as welding, heat treating, cleaning, nondestructive examinations, and analytical evaluations are carried out by qualified personnel and under controlled conditions, in accordance with applicable codes, standards, and specifications, and other special requirements. The qualifications of personnel performing special processes should comply with the requirements of applicable codes and standards. If no such codes or standards exist, the requirements for personnel qualifications should be defined and documented.

14.4.10 Control of Inspections and Tests

Inspections and tests to verify that an item or operation conforms to specified requirements should be planned and documented. The characteristics to be inspected or tested, the methods of inspection or testing, and the criteria for evaluating the results and documenting whether the item or operation is acceptable should be identified.

Records of inspections and tests should include the identity of the item or operation involved, the date, the name of the inspector, the type of inspection or test given, and the results.

14.4.11 Control of Measuring and Test Equipment

Tools, gauges, instruments, and other measuring and test equipment used for operations affecting quality should be controlled to ensure that they meet the defined specifications, are used as designed, and provide the necessary quality of measurement and test data. Measuring and testing equipment should be of the type, range, accuracy, and tolerance needed to accomplish the function intended. At prescribed intervals, or before its use, or whenever its accuracy is suspect, measuring and test equipment should be calibrated and adjusted against certified equipment that has known valid relationships to nationally recognized standards.

Devices that are out of adjustment should be tagged or segregated and not used until they have been recalibrated. Equipment should be properly handled and stored to maintain its accuracy. Records should be kept and equipment should be suitably marked to indicate its calibration status.

14.4.12 Handling, Storage, and Shipment

To prevent damage or deterioration of material and equipment, measures should be established for their handling, storage, shipping, cleaning, and preservation in accordance with work and inspection instructions. When necessary for particular products, special protective environments such as an inert-gas atmosphere, specific temperature levels, absorbent material, and shielding should be specified and provided.

Instructions for marking and labeling items for packaging, shipping, handling, and storage should be established. Any need for special environments or controls should be indicated on the label.

14.4.13 Inspection, Test, and Operating Status

Measures for identifying the inspection and test status of equipment should be established and documented. The status should be known throughout the manufacturing, installation, and operation of the equipment. The inspection and test status should be maintained through the use of status indicators such as physical location, tags, markings, stamps, or inspection and test records. Only items that have passed the required inspections should be installed or operated.

Procedures should be developed to ensure that operations are conducted in accordance with applicable documented instructions and procedures and that items perform satisfactorily in service. Measures such as tagging should also be used to indicate the operating status of systems and components, and to prevent any inadvertent, unplanned use of equipment.

The emergency response capability of the facility (personnel and equipment) should be included in this program. Annual testing of emergency response must be conducted and adequate performance verified.

14.4.14 Nonconformance and Corrective Action

Controls should be established to ensure that failures, malfunctions, and defects in equipment and nonconformances to procedures and processes are promptly identified and corrected. In the case of a significant problem, the controls should ensure that the cause of the problem is determined and that corrective action is taken. The problem, its cause, and the corrective actions needed should be documented and reported to appropriate levels of management. Follow-up action should be taken to verify implementation of corrective action.

Items that do not conform to requirements should be controlled to prevent their inadvertent use. Control provisions should include identifying and disposing of the items and notifying affected organizations.

14.4.15 Quality Assurance Records

Records should be kept identifying operations that affect quality and showing that regulatory and license requirements have been met. The records should be legible, identifiable, and retrievable, and should be protected against damage, deterioration, or loss. Quality assurance records should be centrally maintained by the individual or organization assigned the responsibility for the quality assurance program. Alternate designees may be acceptable. However, in all programs, requirements and responsibilities for record transmittal, distribution, retention, maintenance, and disposition should be established and documented.

The types of records needed for verification of the radiation protection program include radiation exposure records, bioassay data, radiation and contamination survey reports, calibration records, and training records. A more complete listing of required radiation protection records is located in ANSI N13.6-1972.

14.4.16 Audits

Audits by personnel responsible for quality assurance should be scheduled periodically (depending on the importance of the activity being audited) to verify compliance with all aspects of the quality assurance program and to determine the effectiveness of the program. Trained auditors who are not

directly responsible for the areas being audited should follow written procedures or checklists. Audit results should be documented and reviewed by responsible management, and any necessary follow-up action should be taken.

REFERENCES

American National Standards Institute (ANSI). 1972. Practice for Occupational Radiation Exposure Records Systems. ANSI N13.6-1966, Rev. 1972, New York.

CHAPTER 15. APPRAISAL OF RADIATION PROTECTION PROGRAMS

FIGURES

CHAPTER 15. APPRAISAL OF RADIATION PROTECTION PROGRAMS

A variety of methods can be used to ensure that a radiation protection program is providing a reasonable degree of safety. One method, a quality assurance program, was discussed earlier. This chapter focuses on the use of a comprehensive appraisal.

An appraisal is a means of comprehensively evaluating the overall adequacy and effectiveness of the radiation protection program. Unlike a compliance inspection, which is an evaluation of a program by discrete subject areas, an appraisal is an integrated look at the total program. That is, it looks at the total program needs, not just at regulatory compliance. It is focused on identifying and correcting the underlying causes of deficiencies rather than on identifying failures to follow specific procedures or regulatory requirements.

The routine appraisal of a radiation protection program entails verifying that the program is effective in protecting personnel, property, and the environment. This goal can be accomplished through a thorough, technical health physics appraisal by experts, and follow-up by management to ensure that any problems found during the appraisal have been corrected and that staff members are being protected.

This chapter includes a brief overview of the steps for conducting a technical appraisal; a discussion of the areas of a radiation protection program that should be included in an appraisal; a checklist of questions for use during appraisals; and an introduction to network techniques that can be used to help plan an appraisal.

Section 15.1 CONDUCTING A TECHNICAL APPRAISAL

A thorough technical appraisal usually begins with the selection of a team of individuals who are familiar with the requirements of a health physics program and with applicable standards and regulations, and who have the ability

to conduct appraisals. The team members review site documents and then conduct an onsite appraisal that includes discussions with personnel, observation of work practices, and reviews of procedures and reports. Their findings are then reported in writing.

15.1.1 The Appraisal Team

In order to characterize the radiation protection program and identify any deficiencies, it may be necessary to expand the onsite staff with additional personnel who are experts in the field of health physics or to bring in a team of outside consultants who have experience in broad-based health physics programs and who have conducted appraisals. This expanded-team approach, using outside expertise, provides objective viewpoints and reduces the time required for the technical appraisal. In addition, a team approach allows team members with varying backgrounds to interact as they investigate deficiencies and recommend solutions. Their interactions and discussions can help identify problem areas and clarify the causes of symptomatic deficiencies.

The members of the appraisal team should be selected based on the type and size of the radiation protection program to be evaluated. Each team member should have both a broad and thorough knowledge of health physics, and an area of expertise that complements those of the rest of the team. The appraisers should be familiar with current standards, regulatory guides, and regulations, and should have shown through prior appraisal experience that they have an aptitude for conducting appraisals.

A leader of the appraisal team should be selected. This individual disseminates documents, briefs the installation commander, assigns areas of responsibility to other team members, and functions as the team coordinater to ensure that all areas are covered.

15.1.2 The Appraisal Process

The appraisal process begins with a thorough review of site documents, which are distributed to the appropriate team members by the appraisal team leader. These documents should include: 1) the operating license, 2) the environmental impact statement or environmental analysis, 3) program objectives, 4) related missions, 5) organizational charts, 6) job descriptions,

7) performance objectives, 8) training records, 9) work utilization, scheduling, and budget documents, 10) radiation safety manuals, 11) health physics procedures, 12) chemistry procedures, 13) respiratory protection programs, 14) applicable regulations, 15) the emergency plan, 16) procedures for implementing the emergency plan, 17) dosimetry records, 18) survey records, 19) minutes of meetings of the radiation protection committee, 20) reports on previous inspections and appraisals, and 21) any other documents needed to complete the appraisal. During the review and preparation period, each team member should:

1. review the documents received from the team leader to identify tasks that are crucial to detecting and assessing radiation levels, notifying appropriate staff and officials, and implementing protective action
2. identify the individuals responsible for crucial tasks
3. identify the minimum equipment, procedures, and instruments required for the performance of those tasks
4. identify any deficiencies in standard operating procedures (SOPs)
5. identify any deficiencies in emergency plans and procedures.

The time planned for the appraisal should be long enough to allow the team to talk with installation personnel and radiation workers, review and observe work practices, and review onsite radiation protection procedures and records relating to exposures, incidents, etc. The appraisal team should also meet with the installation commander and any other managers between the radiation protection staff and the commander, to ensure that the radiation protection staff has sufficient support to carry out the routine ALARA program (keeping exposures as low as is reasonably achievable) and to handle any abnormal occurrences.

15.1.3 Report of Appraisal Findings

At the completion of the appraisal, a report should be written specifying whether each major component of the radiation protection program was found to be adequate. The total program should also be rated as acceptable, adequate for present operations but having significant weaknesses, or not acceptable.

Deficiencies or weaknesses are considered significant when they have a direct effect on the level of protection provided or when they play a critical part in whether a portion of the program is judged acceptable. For example, failure to calibrate instruments or provide adequate dosimetry would be a significant deficiency that would make all or part of the program unacceptable depending on the necessity of the devices to the overall safety of the program. Isolated weaknesses and minor problems should not be judged as representing a significant finding. However, if a number of deficiencies are found within a particular phase of the program, then an assessment that significant problems exist may be warranted for that phase. If a deficiency or weakness requires immediate attention, the problem should be discussed with the Radiation Protection Officer (RPO) and the cognizant manager, and an immediate solution should be agreed on.

Section 15.2 PROGRAM AREAS THAT SHOULD BE APPRAISED

The elements that make up an effective radiation protection program are the radiation protection organization, the selection and training of personnel, survey programs, programs for the control of internal and external exposure, the ALARA program, facilities and equipment, waste management, and records and audits. Some of the aspects of each area that should be covered both in management reviews and in technical appraisals by health physics experts (members of the appraisal team) are discussed below.

15.2.1 The Radiation Protection Organization

The appraisal of a health physics program begins with an evaluation of the radiation protection organization. Both onsite and offsite support for the radiation protection program should be reviewed. For example, if the Ballistics Research Laboratory has an agreement with ARRADCOM, Dover, New Jersey, or with the Material Test Directorate, Aberdeen, Maryland, to provide health physics support either routinely or during emergencies, then the appraisers should ensure that the supporting organization is aware of the

magnitude of support needed by the requesting organization. The RPO's management should ensure that there are written agreements delineating responsibilities. Additionally, if offpost use of radionuclides is authorized, the appraisers should review the procedures and licenses involved to ensure that they are adequate.

To ensure awareness of responsibilities, an organizational chart depicting the onsite and offsite radiation protection organization, together with the total command, should be available to everyone. This chart should clearly show that the RPO has a direct reporting chain to the base commander. The purpose of this direct access is to ensure the authority to stop work in the event of potential or actual hazardous situations. A written statement of the duties, authorities, and responsibilities of the RPO and the radiation protection staff should also be available. If contractors and private organizations (e.g., fire departments or hospital emergency staff) provide technical assistance to and augmentation of the emergency organization, they should be specified and their roles clearly defined.

Within the radiation protection organization itself, authorities and responsibilities should be clearly assigned. Job descriptions are frequently useful in delineating the scope of responsibilities and ensuring a thorough transition during staff turnovers. The appraisers should also ensure that the radiation protection staff feel they have the authority to implement the radiation protection program. The management review would include a check to ensure that the RPO and the staff have job descriptions, are aware of their responsibilities, and are fulfilling those responsibilities.

The responsibility for preparing emergency plans and procedures is frequently assigned to an individual, in addition to his or her primary duties, without any allocation of the authority, manpower, time, or money needed to accomplish the task. Because the emergency planning program involves a number of persons and organizations, the extent of emergency planning necessary at each site should be carefully evaluated, and the organizations participating in the planning should be aware of who in the radiation protection organization is responsible for coordinating the program.

The appraisers should ensure that the staff of managers, supervisors, and radiation workers is adequate for the amount of radiation work performed at the site, both for operations during the day and for operations after normal working hours. There should be enough radiation protection technicians to perform assigned responsibilities for routine operations, and at installations with a large radiation work force, the staff of radiation protection technicians should include specialists in such areas as dosimetry, respiratory protection, and ALARA review. The technical support personnel should be relieved from clerical duties as much as possible by administrative support personnel, especially during emergencies. There should be emergency plans for supplementing the radiation protection staff within 18 hours of a major accident. This procedure will reduce the potential for mistakes caused by fatigue.

15.2.2 The Selection and Training of Personnel

The quality of the radiation protection program depends on the qualifications of the RPO and on the support the RPO receives from management and the staff. During an appraisal, therefore, the appraisers should review the criteria used to select the site's RPO and radiation protection staff, verify that the RPO and the staff meet these criteria, and assess the programs used to train personnel.

The routine management review should include verification that there are job descriptions for the RPO and the radiation protection staff. These descriptions should be discussed with the individuals to ensure that they are up-to-date and accurately reflect the current work assignments. In conjunction with the work assignment, emergency and routine training should be reviewed. This review would include verification that:

1. training classes are scheduled
2. training is provided as specified
3. radiation workers receive annual training
4. training records are up-to-date.

A. Selection of the Radiation Protection Staff. The criteria used in selecting a site's RPO should be based on the type of work conducted at the installation and the size and type of radiation program involved. However, in all instances, the qualification criteria should include consideration of the

individual's formal education, continuing education, work experience, previous management experience, and technical understanding of health physics. The individual selected should have demonstrated experience in the area that he or she is to manage.

The RPO should be responsible for developing selection criteria for each position in the radiation protection organization and for selecting the technicians who help run the program. The appraisers should ascertain whether the selection criteria are related to the individual jobs and whether they include an assessment of formal education and experience. These criteria should be used for both hiring and promotions, and the staff should be aware of the promotion requirements.

B. Routine Training Programs. The members of the appraisal team who are responsible for appraising a site's training program should have considerable experience in radiation protection training. This experience is necessary in determining whether the training provided is adequate in content, nature, and length. The training must be assessed against 10 CFR 19, against the training requirements for and the complexity of a program, and against the authority for the program. Consideration must be given to the type of work authorized for and conducted on the site.

The training program should be assessed in two parts: training for radiation workers and other staff members, such as medical personnel, public information officers, and security support staff, and training for the radiation protection staff. Training for both groups should include the following:

1. a defined scope and written content for the program
2. instructors qualified in the subjects they are teaching
3. instruction schedules and lesson plans
4. objectives for trainee performance
5. demonstration of standards attained by trainees
6. frequency of required attendance
7. documentation of attendance (including test results, dates, subjects, etc.).

Formal on-the-job retraining should be provided periodically for all individuals.

The following topics should be included in each training program:

1. the specific duties and responsibilities of those being trained
2. the site's reporting or communications chain
3. site-specific or job-specific hazards
4. industrial and radiation safety
5. special procedures
6. special protection (e.g., the use of respirators and protective clothing)
7. the ALARA philosophy.

Training should include instruction in the capabilities and limitations of any instruments to be used. Special procedures and the reasons the procedures are needed should be written down and explained to everyone involved.

An adequate training program should not consist solely of classroom instruction, demonstrations of equipment to the group, and the use of maps or building drawings to point out the location of equipment, work stations, or emergency response duty stations. Rather, training should include hands-on use of equipment and tours of areas that the trainees may need to enter in the course of their work.

The individuals evaluating the training program should attend the training classes to verify the level of instruction. Their evaluation should also include a thorough review of class records for the previous 2 years, discussions with randomly selected individuals to verify that they received and understood the training shown in their records, evaluation of the training aids used, and discussions with the instructors, the supervisors of radiation workers, the radiation protection staff, and the RPO. In evaluating training for the radiation protection staff, the appaisers should also ascertain whether the operators of counting and analysis systems are qualified to operate them and are using them properly. The appraisers should verify that

when new instrumentation is put into use, the staff is retrained in the state of the art for that instrumentation and the range of its capabilities.

C. Emergency Preparedness Training. Training the staff, especially the radiation protection staff, for emergencies is extremely important because emergency situations precipitate changes in reporting chains, the scope and nature of duties, and the perceptions of individuals. Individuals under stress may revert to established behavior patterns; training can help establish patterns appropriate to emergencies and eliminate the randomness of purpose that is characteristic of such situations. The appraisal of the emergency preparedness training program should involve ascertaining whether individuals will respond appropriately when under stress.

The emergency preparedness training program should contain provisions for training the members of support organizations (e.g., the fire department and ambulance service). The purpose of the training should be to ensure mutual understanding of roles, procedures, and interfaces. Although the command cannot always require offsite groups to participate in training sessions, the appraisers should assess the capabilities of these groups to support the RPO and the radiation protection staff in emergencies.

15.2.3 The Radiation Survey Program

The purpose of the radiation survey program is to evaluate actual or potential radiation hazards at facilities where radiation sources are used. The scope of survey activities should be clearly stated for all installations.

The primary emphasis of management's review of the survey program should be verification that survey procedures exist in written form and that the procedures are followed when surveys are conducted. Records should contain the latest surveys and include all required information. The instrument storage facility, the general condition of the instruments, the condition of the emergency kit, the records showing dates for instrument calibration, whether the dates are being met, and whether outdated instruments are being used should also be reviewed.

A. Responsibilities and Scope of the Program. The RPO should be responsible for the design, development, and maintenance of the survey program and should ensure that there are procedures for performing routine and periodic surveys of airborne and surface contamination. The extent of the surveys should be consistent with the hazards and work conditions at the installation. If any special or unusually complex surveys are performed by an offsite team or consultant, the RPO should ensure that the agreement for this work is well defined and should specify the individual or individuals responsible for monitoring the work.

The appraisers should ensure that the RPO has adequately defined the scope of the survey program to include all potential radiological hazards at the installation. They should review the scope and the frequency of the survey routines to ensure that they are adequate for the needs of the program and consistent with regulatory requirements. The appraisers should also determine whether the radiation protection staff and/or the RPO review the routine and periodic survey data and assess the need for possible additional actions.

B. Instrumentation Suitability and Use. The appraisers should determine whether the instrumentation used in the survey program meets the minimum standards required by regulations and the site's license. The instrument technician or RPO should be required to demonstrate that the quantity, type, range, and sensitivity of portable instruments are sufficient for the scope of routine and nonroutine health physics activities. Instruments should provide the type of measurements required for the program.

The appraisers should evaluate the calibration program using reference documents such as Standard N323-1978 of the American National Standards Institute (ANSI). This standard is a general document on instrument testing and calibration that contains extensive technical information. It includes functional testing criteria and calibration methods, and specifies sources, calibration facilities, calibration frequencies, and required records. All calibration sources should be traceable to the National Bureau of Standards (NBS).

The RPO should be responsible for ensuring that a thorough evaluation of the best location is made before a fixed or semifixed instrument is set up. These instruments should be positioned to allow for ease in operational checks, calibrations, and maintenance.

Monitoring for airborne radioactive materials should involve the use of breathing-zone samplers, area air samplers, portable and semiportable air samplers, and grab air samples. The appraisers should observe several air samples being taken and should review air-sampling records to ensure that the proper procedures are being followed and that the air samples taken are representative of the air being breathed by workers. If the persons taking the samples fail to consider air currents and the dilution and turbulence caused by work activities, the samples taken may not represent the air being breathed.

The appraisers should determine whether emergency kits and survey instruments have been placed at appropriate locations. If so, the instruments should be evaluated to determine their suitability for each location.

The RPO should be responsible for specifying the methods and equipment to be used for routine surveys of offsite locations and for emergency offsite radiological surveys. For all onsite and offsite surveys, each member of the survey team should be required to record the following information:

1. date and time of each survey
2. location of each survey
3. name(s) of the individual(s) who performed the survey
4. the instrument used, identified by type and serial number
5. the mode in which the instrument was used (i.e., window open or closed)
6. the duration of the meter or instrument reading
7. air sampler flow rates
8. background radiation levels at the time of air sample counting
9. sample count time
10. work condition at the time of sampling.

C. Records. The appraisers should ensure that the RPO verifies the documentation of all surveys. Survey reports should be clearly written and traceable as to instrument, date, time, location, and project. The records of SOPs should correctly reflect the job and work conditions. The appraisers

should thoroughly review the records to determine whether the RPO ensures that survey results are distributed to staff members and supervisors as necessary.

15.2.4 The Program for Internal-Exposure Control

Management review of the program for internal-exposure control requires both a walk-through and a review of records. The purpose of the walk-through is to ensure proper posting; proper cleanup of contaminated areas; proper storage of respirators (individually wrapped and stored in a closed container); proper wearing of respirators; storage of radiaoctive liquids in nonbreakable containers or in locked storage containers; proper wearing of lapel air samplers; proper positioning of breathing-zone samplers; and proper storage of air-monitoring equipment. In addition to the walk-through reviews, management should ensure that internal-dosimetry records are maintained for everyone who has received an internal dose, is suspected of having received an internal dose, or has entered an area containing airborne radioactivity, whether with or without a respirator. The manager should also review individual dose records to ensure that they are up-to-date; calibration records for bioassay, air-monitoring, and air-sampling equipment to ensure that they are up-to-date; the RPO's trend analysis for indication of increased activity; all operations to ensure that there are procedures for each; and the RPO's records of surveys versus maximum permissible concentration-hours (MPC-hrs) to ensure that the RPO has a method for interpreting whole-body-counting data that relate to the working environment.

Individuals working with radioactive materials may work with unencapsulated sources in physical forms or in chemical solutions. When these materials are unintentionally released from their containers, they can be inhaled, ingested, or absorbed through the skin. Therefore, radiation protection programs for such workers should include 1) methods to limit internal exposures, 2) an internal-dosimetry program, 3) reviews of exposures, their causes, and the corrective actions taken, and 4) a quality assurance program. All of these aspects of the program for internal-exposure control should be reviewed during the technical appraisal.

A. Exposure Limitation Methods. Two important methods of limiting internal exposures are administrative and engineered safeguards.

(1) Administrative Safeguards. The administrative approach to controlling internal exposures usually consists of site-assigned dose limits and written procedures. The limits should be considered in the establishment of procedural and physical controls. The written procedures should be well disseminated and read by all radiation workers and support personnel who may have reason to go into an area of potential airborne radioactivity. The appraisers should ensure that the procedures define clearly when protective clothing and equipment are needed and include a means of ensuring that only qualified personnel use respirators.

The procedures should define the requirements for posting controlled-access areas and areas where airborne or other contamination is known to exist. The appraisers should ensure that suitable measures are taken to minimize leakage, control local releases, and clean up contaminated areas, and that there are adequate plans for expanding the respiratory protection program in the event of an accident.

(2) Engineered Safeguards. Engineered safeguards against internal exposure are provided by containment and ventilation systems, contamination control, alarm systems, and respirators. The reviewers appraising the control of internal exposure should begin by reviewing the first three of these items to ensure that every effort has been made to minimize the number and size of areas containing airborne radioactivity.

Respirators are the primary physical device for minimizing internal exposures. The use of respirators in either an NRC-approved or a nonapproved program to reduce the potential for inhalation of radioactive material constitutes a respiratory protection program. The commitment to a quality respiratory protection program should begin with a written policy statement on respirator usage issued by a high management level (e.g., by the installation commander). This policy statement should discuss the objectives of the program and assign the responsibility for its operation to the RPO.

The issuance, maintenance, and repair of respirators, and the training of personnel for their use, should meet the guidelines found in such documents as the Nuclear Regulatory Commission's NUREG-0041 (NRC 1976). Before beginning the appraisal of the respiratory protection program, the appraisers should

ensure that they are intimately familiar with these support documents. The degree to which NUREG-0041 is applied at a site will depend on the peculiarities of the individual program. However, the appraisers should ensure that every program includes at least the following items: medical examination of each respirator user by a qualified physician, including pulmonary measurements; training in proper respirator use; use of only those respirators approved by the National Institute of Occupational Safety and Health (NIOSH); an inspection program to ensure that breathing air meets the requirements of ANSI Z88.2 (1980) and the Occupational Safety and Health Administration (OSHA); and a program for cleaning and maintaining respirators for both hygienic and contamination control purposes.

The individuals responsible for training radiation workers in the proper use of respirators should have received their training directly from a certified respirator manufacturer. Their training should have included proper fitting of masks and repair procedures.

B. Dosimetry Program. An internal-dosimetry program consists of measurements of the concentration of airborne radioactive materials in the workplace; bioassay measurements, for estimating the quantity of radioactive materials deposited in various body organs; measurements for determining ionizing-radiation doses to body organs; and techniques for assessing these measurements.

The appraisers should determine whether the bioassay techniques and counting facilities used at an installation are sufficient to permit a reasonable assessment of the internal burdens of the radionuclides used at that installation. The bioassay techniques should include the use of models or calibrations to ensure the accuracy and reproducibility of measured findings. The operating manual at each site should state, for each technique used, the type of radiation detectable by the technique, the sensitivity and accuracy of the system, the calibration sources used and the activity and intensity of each, and whether the system is sensitive enough to detect a concentration equivalent to 5% of the maximum permissible body burden (MPBB) for the most restrictive radionuclide in a mixture of radionuclides. This determination must be within a 95% confidence level.

Appraising an internal-dosimetry program is a detailed and complex process that requires the knowledge of an expert with many years of practical experience. The appraiser needs to use reference documents such as ANSI N343-1978 and Publication 2 (1959) of the International Commission on Radiological Protection (ICRP).

C. Exposure Review. The appraisers should determine whether radiation exposure and/or dose limits for routine operations and nonroutine events are maintained ALARA and whether survey and internal-exposure data on individuals are routinely compared with each other and with the limits. When the limits are exceeded or closely approached, the appraisers should determine whether the RPO takes corrective action and how effective that action is. To support the RPO, managers, supervisors, and foremen of operations and support groups should strive to keep both individual and group exposures, and the number of workers exposed, at a minimum. The existence of SOPs that require the signature of the RPO or a designee, the radiation worker, and the worker's supervisor is a good indicator of such an effort.

D. Quality Assurance Program for Internal Dosimetry. The primary purpose of the quality assurance program is to ensure that the data gathered in the internal-dosimetry program represent the best efforts possible in dose assessment. To this end, the RPO should establish calibration frequencies for, and the quality assurance staff should review, each dosimetry system and dose assessment technique. The appraisers should ascertain whether the RPO periodically evaluates the quality assurance and calibration reports to determine whether the established calibration frequency is adequate for each system used.

The appraisers should ensure that whole-body counting equipment is calibrated using sources traceable to the NBS. These sources should cover the entire spectrum of radionuclides currently in use at the installation and should vary in strength from the lower limit of detection of the counting system to realistic accident levels. The appraisers should also determine whether the whole-body counting system is calibrated at least annually. The routine calibration program should include an interim calibration with tolerance limits that, if exceeded, require recalibration of the entire system.

Guidance on the calibration of whole-body counting systems can be found in documents such as ANSI N343-1978.

The appraisers should determine whether the installation has a procedure for estimating MPC-hr exposures from whole-body-counting data. Because 10 CFR 20.103 expresses standards for internal emitters in terms of time-integrated concentrations (MPC-hrs) and intakes rather than permissible body burdens or doses, it is important that the RPO 1) maintain a comprehensive breathing-zone air-sampling program, and 2) be able to compare whole-body or organ burden data with the data generated by the air-sampling program. To accomplish this, the RPO must have a method for interpreting whole-body-counting data in terms of the MPC-hrs of exposure needed to produce the measured burden. One of the main reasons for relating the data base on whole-body counting to the data base on air sampling is to determine the effectiveness of the respiratory protection program.

15.2.5 The Program for External-Exposure Control

Management review of the program for controlling external exposure, like that of the internal-exposure program, involves observation of work practices and review of records. Management can perform an informal, walk-through review by being aware of whether radiation areas are posted properly, dosimeters are worn properly where they are required, radioactive waste is stored properly, and waste containers are labeled. Managers who are not familiar with proper procedures in these areas can consult AR 40-14, AR 385-11, and the RPO. In addition to the walk-through reviews, management should ensure that dosimetry records are maintained for everyone issued a dosimeter; that supervisors use dosimeter results when planning jobs and staff assignments; that all personnel are given the results of their annual dosimeter reading; that the RPO knows the procedure for reporting an overexposure; and that a sufficient number of dosimeters are available for routine and emergency use as well as for visitors.

The technical appraisal of the external-exposure control program should include review of 1) the methods used to limit exposures, 2) the dosimetry program, 3) the reasons for exposures and any corrective actions taken, and 4) the quality assurance program.

A. Exposure Limitation Methods. Both administrative and engineered safeguards should be reviewed as part of the appraisal. The administrative means of controlling external exposures usually consist of site-assigned dose limits and written procedures for minimizing exposures. The appraisers should review the dose limits to determine their usefulness and the ability of the staff to meet them. They should also talk with randomly selected radiation workers to verify their awareness of the administrative guidelines. These guidelines should clearly reflect existing regulations and recognize the ALARA concept.

The use of physical barriers for exposure control should be reviewed by the RPO on a regular basis and the results should be documented. The appraisers should evaluate the use of barriers and talk with radiation workers to determine their effectiveness. If remote-operating and remote-handling devices are available, the appraisers should ensure that the individuals authorized to use them have received special training and that the devices are well maintained. In areas with access alarms, periodic tests of the alarms should be performed to ensure their operation, and placards showing the potential hazards of the areas should be clearly displayed.

B. Dosimetry Program. Before beginning this phase of the appraisal, the appraisers should assure themselves that they are familiar with the current standards in the area of external dosimetry, including ANSI N13.11-1980.

The appraisers should review all sources licensed for use at the installation to ensure that the dosimetry program is suitable for the types and levels of radiation exposure anticipated during routine and nonroutine work. They should also evaluate the personnel responsible for the dosimetry function to determine whether they have adequate knowledge to perform routine duties and to recognize unusual events that may require special interpretations or evaluations. Appraisals frequently reveal that the readings from film or thermoluminescence dosimeters are not compared with the readings from secondary dosimeters (e.g., pocket ionization chambers). When comparisons are made, an acceptance criterion, or level at which follow-up action is required, should be specified.

An installation that sends dosimeters offsite for calibration and/or processing should have a quality assurance program. This program should include the use of spiked dosimeters and blanks. Secondary dosimeters should be carefully screened before they are put into service. The acceptance of offsite work without an independent quality control check represents a failure by the RPO to take responsibility for the accuracy of the dosimetry program.

The appraisers should ensure that the equipment and facilities available are adequate for nonroutine dosimetry and exposure control. Enough dosimeters of acceptable quality and sensitivity should be available for short-term use by personnel or visitors to areas requiring dosimeters. Exposure records should be kept current and should be sent to workers and their supervisors frequently and promptly enough to ensure their usefulness.

C. Exposure Review. The appraisers should determine whether the exposure data generated by dosimeters and instruments are routinely reviewed by management and whether any discrepancies between the primary and secondary dosimeter readings that exceed the acceptance criterion are followed up by an investigation of the exposure conditions.

The RPO should maintain a plot of exposures that shows trends and indicates whether doses are being kept ALARA. These plots can be cross-referenced by job, location, profession, and total work force. The RPO should also have records of each review of the trend plot and the results of that review.

D. Quality Assurance Program for External Dosimetry. The quality assurance organization should play an active role in the program for controlling external exposures. The appraisers should assess the quality assurance functions performed by the RPO and determine whether the quality assurance representative assists in reviewing procedures and ensures that there is suitable feedback from management. For onsite calibration of instruments, devices, and processes, the quality assurance representative should assist in establishing acceptance criteria. The appraisers of the quality assurance program should pay careful attention to the records maintained by the quality assurance office.

15.2.6 The ALARA Program

The appraisers should verify that management has a written policy showing commitment to ALARA and administrative procedures to implement the policy.

The RPO should be responsible for overseeing the ALARA program as described in NRC Regulatory Guide 8.10 (1975). However, an individual in management should be responsible for working with the RPO to ensure that mechanisms for keeping exposures ALARA are instituted at the site.

The appraisers should determine whether an adequate system has been established to avoid unnecessary or inadvertent personnel exposures. Shielding should be used when equipment is being serviced; measures should be taken to provide distance from sources, when possible; and easy access to equipment should be provided. The appraisers should determine whether remote-handling tools or remote readouts are used when necessary. They should also thoroughly review the entire radiation protection program to determine the effectiveness of the ALARA program in reducing exposures.

The appraisers should interview radiation workers to determine their concept of ALARA, whether adequate training, preparation, and planning are incorporated into work activities, whether the radiation protection staff become involved early in the planning of work, and whether a debriefing is held when a job is completed to determine more effective means of reducing exposures.

Management review of this area should be limited to ensuring that there is an ALARA program review committee, that exposure information is used for job planning, and that personnel are familiar with the ALARA principle.

15.2.7 Facilities and Equipment

Management review of facilities and equipment should include a walk-through and visual inspections of equipment. The walk-through should center around the availability of sufficient space for calibrations, sample analysis, and the use of laboratory counters. Management should also inspect the condition of support equipment (e.g., protective clothing). The technical appraisal should cover the topics discussed below.

A. Facilities. At each installation, the appraisers should evaluate whether there are sufficient locations and space for the following: counting room, calibration of instruments, personnel decontamination, access control, offices, equipment decontamination, instrument storage, external dosimetry,

internal dosimetry, the fitting, testing, and cleaning of respirators, training, contaminated-equipment storage, and laundry. When a new facility is being designed, the RPO should be involved in an ALARA review of that structure.

If the installation uses large enough quantities of radioactive material so that the potential for offsite releases is a concern, the appraisers should ensure that the RPO has made provisions for offsite decontamination of personnel and has determined whether local hospitals have sufficient space and equipment to handle emergencies involving contaminated victims.

B. Protective Equipment. Respirators, protective clothing, temporary shielding, and containment materials should all be reviewed as part of the equipment appraisal.

(1) Respirators. The supply of respirators should be adequate for handling routine and abnormal operations. The installation should have an agreement with a commercial company or another command for the rapid procurement of extra respirators and for the expansion of decontamination and repair services in the event of an emergency.

(2) Protective Clothing. Protective clothing should be stored in a number of locations so that all of it is not lost in the event of a fire or accident. The supply should be adequate for handling routine and nonroutine operations. For accident situations, special clothing such as disposable paper and plastic suits should be available. Contamination limits for reusable clothing should be established. When the level of contamination on the clothing exceeds the limit, the clothing should be disposed of.

(3) Temporary Shielding. The appraisers should ensure that an adequate supply of temporary lead shielding, such as bricks, blankets, lead shot, and lead sheets, is available. The radiation protection staff should be trained in the proper use of these supplies and instructed to carefully survey the temporary shielding before removing it to avoid spreading contamination.

(4) Containment Materials. The supply of containment materials (e.g., heavy-gauge plastic sheeting, plastic windows, and nonskid floor covering) should be adequate for handling routine and nonroutine operations. The RPO should carefully analyze these materials for compatibility with the work

environment they will be used in. The site should have detailed procedures on the use of these materials, and the radiation protection staff should be trained in their use.

15.2.8. Management of Radioactive Waste

Appraisal of the waste management program should include a review of records and a walk-through inspection of all areas where waste is either generated or stored. All waste should be stored in authorized, appropriately labeled containers free of exterior contamination, rust, and corrosion. The appraisers should ensure that radioactive waste is collected separately from nonradioactive waste and that it is promptly removed from the generator's location and stored in properly posted areas apart from work locations. Control procedures should be used to minimize personnel exposures.

The appraisers should inspect waste that is ready for transport to determine whether it has been packaged and labeled accoording to Department of Transportation (DOT) and Department of the Army (DA) regulations. The appraisers should verify the availability of suitable packaging material, as well as packaging procedures. Trucks holding waste for transport should be inspected to determine that they are in compliance with DOT and DA regulations.

Waste records should be reviewed to ensure that an inventory of all waste generated and disposed of is maintained. The total quantity of radioactive material disposed of into the sanitary sewage system, the air, and nearby streams as a result of all activities at an installation must not exceed the quantity for a single licensee given in 10 CFR 20. Records for the transport of waste should be reviewed to determine whether they meet DOT and DA regulations.

15.2.9 Records and Audits

The records management system should be reviewed to determine whether records of each component of the radiation protection program are maintained. In addition, the appraisers should review the procedures for records disposition, traceability, retrievability, and physical protection. Audit records should be specifically reviewed to determine whether the program is periodically audited by individuals with the appropriate technical expertise and to verify that all audit findings have been corrected promptly.

Section 15.3 CHECKLIST OF QUESTIONS FOR APPRAISING A RADIATION PROTECTION PROGRAM

A checklist of questions should be developed for use during appraisals. The example checklist presented in this section is not comprehensive, but is intended to provide an overview of the areas of interest in an appraisal, based on the discussion in the preceding section.

15.3.1 The Radiation Protection Organization

1. Is there an organizational chart depicting the installation's interrelationship with the radiation protection organization?
2. Does the base commander have a working relationship with the RPO?
3. Does the organizational chart show that the RPO has a direct reporting chain to the base commander?
4. Does the RPO's manager exhibit a clear understanding of the goals of the radiation protection organization?
5. Is there evidence of strong management commitment to radiation protection (e.g., written policies or administrative procedures)?
6. Is there a clear assignment of authority and responsibility within the radiation protection organization?
7. Does the radiation protection staff have adequate authority to ensure that the radiation protection program is implemented?
8. If classified work is being done, does the RPO have adequate clearance and unfettered access to ensure that the work is being conducted safely?
9. Is there sufficient staffing within the radiation protection organization to provide adequate coverage of all work with radiation?
10. Is the RPO included in the design phase of operations involving radioactive material?
11. Is the RPO or a designee required to authorize all SOPs?
12. Does there appear to be open communication between the RPO and both radiation workers and other staff members?

13. Is there adequate administrative support to relieve technical personnel from clerical duties?

14. Has a management level individual (e.g., the RPO or a higher-level person) been designated the responsibility for emergency preparedness?

15. Are written emergency plans and procedures available that are commensurate with the degree of hazard?

16. Are there established procedures for obtaining offsite support?

15.3.2 The Selection and Training of Personnel

1. Is there a radiation safety training program for staff members commensurate with their responsibilities?

2. Is there a training program for the radiation protection staff?

3. Does training for the general staff members and the radiation protection staff include the following?
   a. a defined scope and written content for the program
   b. instructors qualified in the subjects they are teaching
   c. instruction schedules and lesson plans
   d. objectives for trainee performance
   e. demonstration of standards attained by trainees
   f. frequency of required attendance
   g. documentation of attendance (including test results, dates, and subjects).

4. Is the scope of the training adequate in content, nature, and length?

5. Do training programs include hands-on use of equipment and tours of areas that the trainee may need to enter in the course of work?

6. Are the operators of the various counting and analytical systems properly and adequately trained in the use of the systems?

7. Is formal on-the-job training available at appropriate intervals for all individuals?

8. Are requalification and retraining in the state of the art of instrumentation available for personnel?

9. Is there a documented training program covering emergency preparedness?

10. Are operators of computer-based analysis systems capable of manual calculations in the event of a power loss?

11. Are training records maintained for a minimum of 5 years?

12. Are the training records complete enough so that the quality, duration, location, and content of training can be ascertained?

15.3.3 The Radiation Survey Program

1. Is there a clear definition of and basis for the survey program?

2. Are procedures for performing routine and periodic surveys well defined?

3. Does each survey record contain as a minimum the following?
   a. survey purpose
   b. survey frequency and location
   c. survey technique
   d. instrument selection, calibration, and use
   e. data and records disposition
   f. status of follow-up actions.

4. Do procedures or policy statements delegate to the radiation protection organization the responsibility for reviewing all SOPs?

5. Are the data from routine and periodic surveys reviewed by the RPO for technical content and possible additional action?

6. Are all surveys well documented?

7. Do SOPs correctly reflect job and work conditions?

8. Is there timely and adequate feedback of analytical results to staff personnel?

9. Is the recordkeeping system commensurate with the guidelines outlined in Chapter 13 of this manual and those in ANSI N13.6-1966?

10. Are radiation areas properly posted in accordance with 10 CFR 20.203?

11. Are portable instruments of sufficient number, type, range, and sensitivity available for routine and nonroutine activities?

12. Do instruments have a calibration sticker that specifies the date the instrument should be recalibrated, the name or initials of the person that performed the calibration, the actual calibration date, the source used for calibration, and the location of the calibration facilities?

13. Are air-sampling instruments sufficient in number, sampling range, and type for the scope of routine and nonroutine activities?

14. Are there procedures that specify the calibration frequency for all instruments?

15. Are calibration sources traceable to NBS?

16. Are inoperative instruments properly marked, stored, and repaired?

17. Are instruments dedicated to sample analysis properly maintained?

18. Are instrument dials and scales clearly legible?

19. Are survey results plotted and reviewed for possible trends?

15.3.4 The Program for Internal-Exposure Control

1. Is there a bioassay program commensurate with the level of hazard at the installation?

2. Are baseline whole-body counts or urinanlyses performed on personnel before they begin work with radioactive material?

3. Are the bioassay techniques used at the site based on the radionuclides used there?

4. Are the sensitivities of the bioassay procedures adequate for assessing maximum permissible body burdens and maximum permissible concentrations?

5. Is there a written procedure for correlating air-sampling results and bioassay results?

6. Are internal-dose limits for routine operations and nonroutine events maintained ALARA?

7. Are incidents of personnel contamination documented, and are the causes investigated?

8. Are adequate records maintained on all individuals who have received an internal deposition of radioactive material?

9. Are uptake limits considered in the establishment of administrative and engineered safeguards?

10. Are there procedures that aid in determining the need for protective clothing and equipment?

11. Are there well-defined procedures for posting controlled-access areas and areas where airborne or other contamination is known to exist?

12. Are proper measures taken to minimize leakage, control local releases, and clean up contaminated areas?

13. Are there adequate procedures for preventing or controlling cross-contamination of samples?

14. Are air flows from areas of low to areas of high airborne radioactivity?

15. Has management issued a written policy statement on the use of respirators?

16. Are there methods of ensuring that only qualified personnel use respirators?

17. Does the person responsible for the respiratory protection program have the ability, training, and experience to do the following?
    a. evaluate total hazard
    b. recommend engineering controls
    c. specify appropriate respiratory protection factors and equipment
    d. forbid use of equipment when conditions warrant.

18. Are sufficient records maintained to evaluate the effectiveness of the respiratory protection program?

19. Do the issuance, maintenance, and repair of respirators, and the training of personnel for their use, meet the guidelines found in such documents as NUREG-0041?

20. Do all personnel who wear respirators have documentation of a complete bronchio-pulmonary examination?

21. Are all respirators used at the installation of the type approved by NIOSH?

22. Are there provisions to ensure the proper fit of respirators?

23. Are medical personnel given enough guidance to adequately evaluate the ability of wearers to use the equipment?

24. Are respirators fitted, inspected, tested, and repaired, and are the wearers trained, in accordance with NUREG-0041 or its equivalent?

15.3.5 The Program for External-Exposure Control

1. Is the dosimetry program suitable for the types and levels of radiation exposure anticipated during routine and nonroutine operations?

2. Are there suitable devices, exposure models, and data bases for measuring or calculating extremity exposures?

3. Can skin exposures be determined by modeling or measurement?

4. Are there suitable techniques, devices, or instruments for measuring neutron exposures?

5. Are dosimeters of acceptable quality and sensitivity available for short-term use by personnel or visitors to areas requiring dosimeters?

6. Are dosimeters being worn in the proper position on the body and/or extremities?

7. Are exposure records on all personnel wearing dosimeters kept up-to-date?

8. Are exposure data reviewed routinely by management, and are the reviews documented?

9. Are discrepancies between the readings of primary and secondary dosimeters reviewed by management (RPO or higher levels)?

10. Are exposure results and exposure histories evaluated against the ALARA requirements of AR 40-14 and 10 CFR 20 as part of a routine review?

11. Do administrative procedures clearly establish action levels and required actions in the event of an exposure that exceeds administrative limits?

12. Do procedures clearly reference and reflect existing regulations and recognize and incorporate the ALARA concept?

13. Are there written procedures for the posting of various hazardous or potentially hazardous areas in accordance with 10 CFR 20?

14. Is the RPO thoroughly familiar with the location of all radioactive material used at the installation?

15. Does the dosimetry program include the use of dosimeters spiked with known types and quantities of radiation, to provide a quality assurance check during processing?

16. Are dosimeters stored in a controlled location to reduce adverse environmental effects?

17. Are control dosimeters included in all shipments to the dosimeter processor?

18. Is the RPO responsible for the control, issuance, and evaluation of all dosimeters?

19. Are there routine quality assurance reviews of the dosimetry program?

20. Is quality assurance extended to the review of procedures?

15.3.6 The ALARA Program

1. Is there a written management policy showing commitment to ALARA?

2. Are there written administrative procedures to implement the ALARA policy?

3. Do facility and equipment design features incorporate ALARA concerns?

4. Is work adequately prepared and planned for?

5. Is the radiation protection staff involved in the planning of work?

6. Are formal or informal postoperational briefings held?

7. Are engineered safeguards used to keep exposures ALARA?

8. Is surface contamination controlled adequately?

9. Are remote readouts available?

10. Are unnecessary exposures during routine surveys minimized?

15.3.7 Facilities and Equipment

1. Are there sufficient locations and space for the following: sample counting, calibrations, personnel and equipment decontamination, access control, offices, instrument storage, external and internal dosimetry, fitting, testing, and cleaning of respirators, training, contaminated-equipment storage, and laundry?
2. If a new facility has been designed, was an ALARA review of the structure performed?
3. Are adequate supplies of protective clothing, respirators, temporary shielding, and containment materials available, and is the radiation protection staff trained in their use?
4. Are all radiation areas posted and isolated from controlled areas?
5. Is access to radiation areas controlled?
6. Are sinks, drain lines, and water supplied to a radiation area isolated from the sanitary sewer?
7. Are radiation areas ventilated to prevent the flow of air into uncontrolled areas?
8. Is emergency equipment available (e.g., fire extinguishers, safety showers, telephones)?
9. If a potential for offsite releases exists, have provisions been made for offsite decontamination of personnel, and do local hospitals have sufficient space to handle emergencies involving contaminated patients?

15.3.8 Management of Radioactive Waste

1. Is the use of radioactive material planned so that a minimum of radioactive waste is generated?
2. Is radioactive waste separated from nonradioactive waste?
3. Is waste segregrated by physical form, half-life, and type of nuclide?
4. Are containers used for temporary storage properly labeled, strong, leaktight, and free of exterior contamination, rust, and corrosion?

5. Is radioactive waste stored away from the work area?

6. Are appropriate control procedures used to minimize personnel exposure?

7. Are all waste generation and storage areas monitored to ensure contamination control?

8. Is waste for transport packaged and labeled according to DOT and DA regulations?

9. Is the total quantity of radioactive material disposed of into the sanitary sewage systems, the air, and nearby streams as a result of all activities at the installation less than the quantity for a single licensee given in 10 CFR 20?

15.3.9 Records and Audits

1. Are records maintained for each component of the radiation protection program?

2. Does the records management system include the identification of specific records, the disposition of records (review, storage, retention period), traceability to the originator, retrievability for audits or investigations, provisions for periodic audits, and physical protection for legal records?

3. Are there complete and up-to-date personnel files for all radiation workers?

4. Is DD Form 1141 (or the automated dosimetry records) filed in each individual's personnel file?

5. Are records maintained in accordance with the guidance in Chapter 13 of this manual?

6. Are records maintained in accordance with the guidance in 10 CFR 19 and 10 CFR 20?

7. Is the radiation protection program audited periodically?

8. Does the quality assurance staff conduct performance audits?

9. Are previous audit reports reviewed before new audits are conducted?

10. Are audit findings corrected within a reasonable time?

11. Are technical audits performed by individuals with extensive experience in the areas in question?

Section 15.4 NETWORK TECHNIQUES FOR PLANNING APPRAISALS

The manager or command group that is looking for an appraisal technique to aid in planning a radiation protection appraisal is faced with a bewildering family of network methods. A network is an organized way of thinking about complex problems by using common sense to determine a sequence of logical steps. Managers (such as RPOs) today face a great increase in the complexity of their work; because they are often dealing with the future (limiting future exposure, planning future facilities), they also face uncertainty. Network techniques were designed specifically to deal with the factors of complexity and uncertainty.

15.4.1 The Function of Networks and Logic Trees

The first network method for controlling projects, PERT (Program Evaluation and Review Technique), was developed for use on the Polaris Submarine program by the U.S. Navy in 1958. The second, more successful method was developed by the DuPont Company and is called the Critical Path Method (CPM). A critical path is defined as a sequence of elements of a program that are dependent upon one another. For example, the radiological survey of a waste container is dependent upon the training of the staff members and the proper response of the survey instruments. In turn, the proper response of a survey instrument is dependent upon its calibration, physical condition (whether it is damaged), and power source (strength of batteries). Therefore, an adequate survey is dependent on several critical paths. The critical paths for this particular example are shown in Figure 15.1 by the use of arrows. Although the PERT and CPM networks cannot be used directly in planning appraisals, they demonstrate the value of logically displaying the relationships among the basic elements of a program, and thus they led to the development of more useful methods such as MORT (Management Oversight and Risk Tree Analytical Logic Methodology).

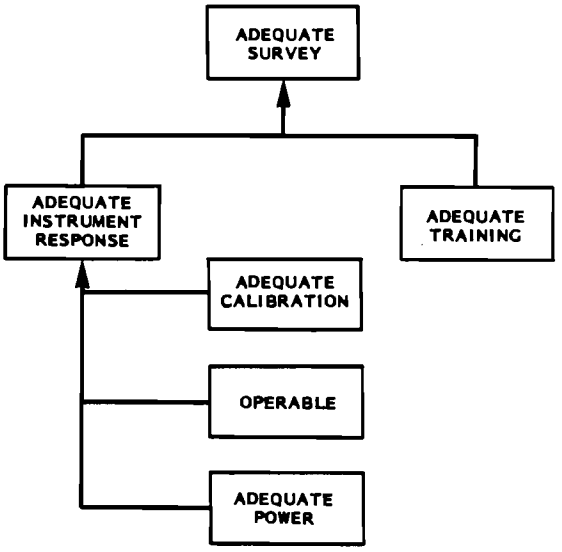


FIGURE 15.1. Critical Paths for an Adequate Survey

The MORT system is a logic tree network that was designed for use in investigating the causes of accidents, or undesirable conditions. However, this system and others like it can be modified to graphically depict a single desirable condition, the starting point for the tree, and systematically proceed through lower levels or tiers until all important factors that produce the desirable condition have been identified. This concept is shown in Figure 15.2, where a tier may be dependent upon several critical paths, as shown by an "AND" gate, or may be dependent upon only one critical path, as shown by an "OR" gate. An example of a logic tree structure is shown in Figure 15.3, where the desirable condition is for a process to be operationally ready.

An appraisal program developed using logic trees would be broken down into many branches, each specific to a single desirable condition or set of related conditions. Each branch would have some point of interface with at least one other branch or tree. The interfaces between branches or trees are important in the evaluation process: data collected from the appraisal of one area must be transferred to another area and considered in the evaluation of both. Through this process, the impact of a particular finding can be assessed in a systematic way, with a minimum expenditure of time and effort. The examples of logic trees presented in this section are a combination of several

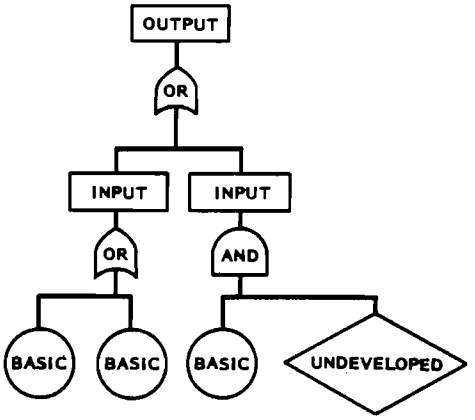


FIGURE 15.2. MORT Logic Tree

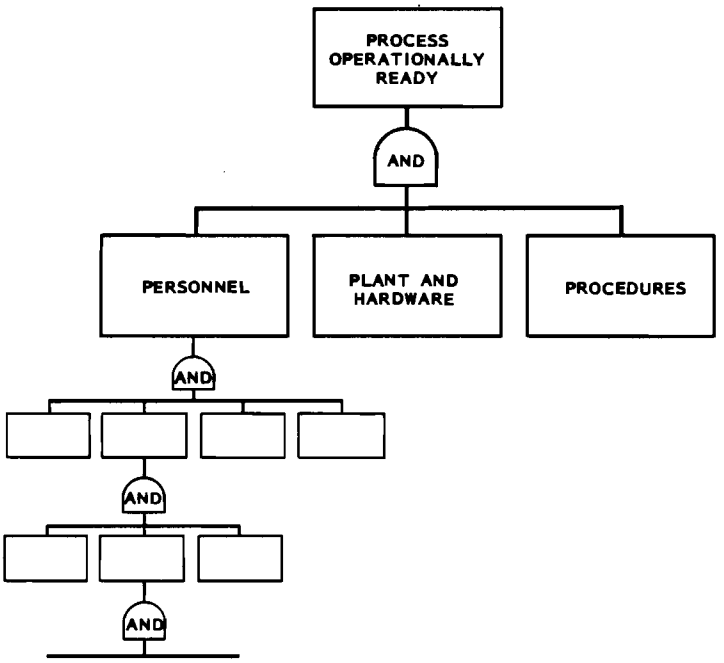


FIGURE 15.3. Logic Tree for a Process To Be Operationally Ready

network systems. They are intended not as an all-inclusive listing of factors related to a radiation protection appraisal, but rather as a means of acquainting the reader with the system of logic trees and their use in developing an appraisal program.

15.4.2 Using Logic Trees to Plan a Radiation Protection Appraisal

The first step in developing an appraisal program is to establish the objectives of the radiation protection program (e.g., to keep exposures as low as is reasonably achievable (ALARA) and to minimize the potential for accidental exposures). On the diagram of the logic tree, this objective is placed in a box that becomes the goal of the total logic tree (see Figure 15.4). Accomplishing such an objective requires both a routine operation and an emergency operation. This dual requirement is shown by the "AND" gate beneath the top box in Figure 15.4. Figure 15.5 shows the further subdivision of an effective routine program into its major components. A deficiency in any of these components could cause the entire routine program to be inadequate. Therefore, an "AND" gate is used to show the relationship of the routine program to its major components. The emergency operation, however, can be satisfied by either a modified routine operation or a special emergency program. Therefore, the emergency operation diagrammed in Figure 15.6 has an "OR" gate to show its relationship to its components. The combination of Figures 15.4, 15.5, and 15.6 yields the logic tree structure for the first two tiers of the radiation protection program, as shown in Figure 15.7.

In a complete appraisal program developed using logic trees, each of the components of a routine program would be further subdivided from two to eight times. The subdivisions of the component "Internal-Exposure Controls" are shown in Figure 15.8. The degree of subdivision into lower tiers depends on the complexity of the radiation protection program. A recent appraisal of the radiation protection programs at operating power reactors involved the use of

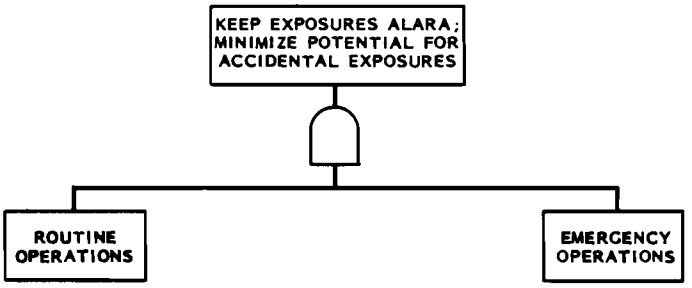


FIGURE 15.4. Radiation Protection Program, First Tier

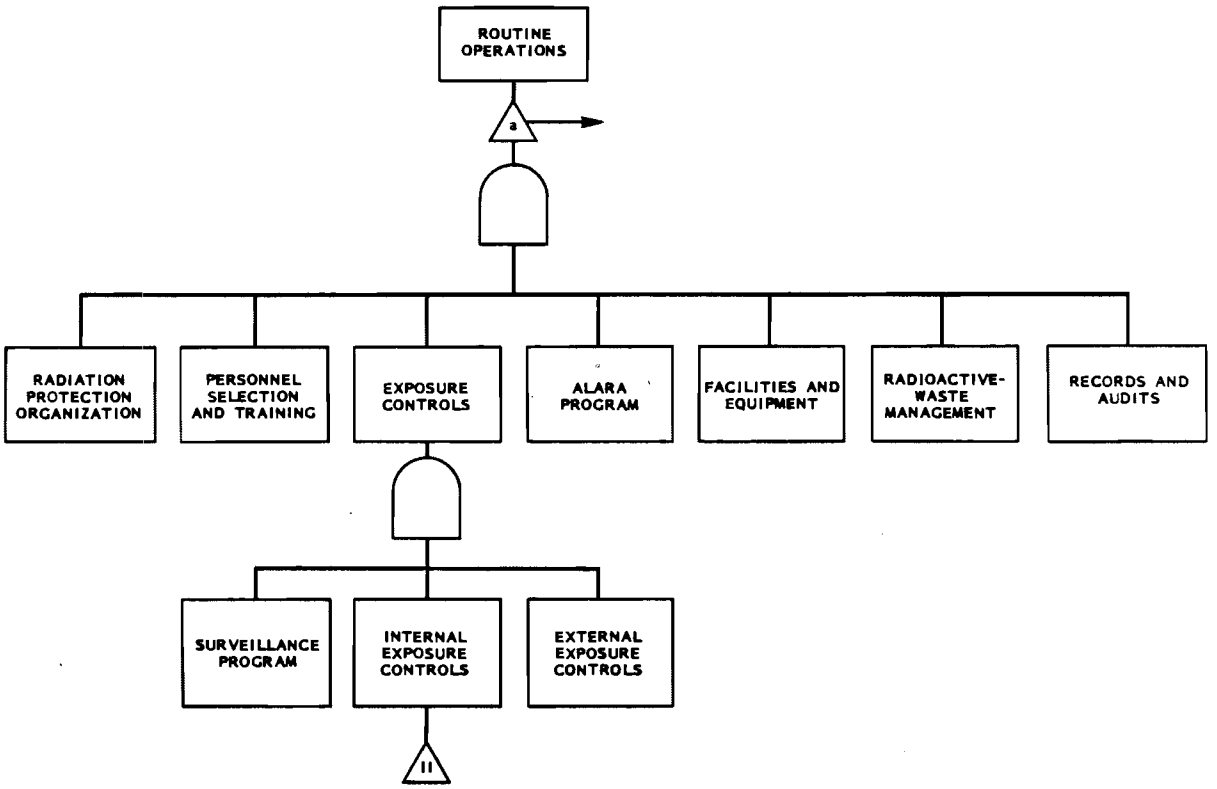


FIGURE 15.5. Radiation Protection Program, Second Tier--Routine Operations

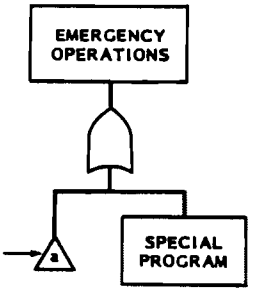


FIGURE 15.6. Radiation Protection Program, Second Tier--Emergency Operations

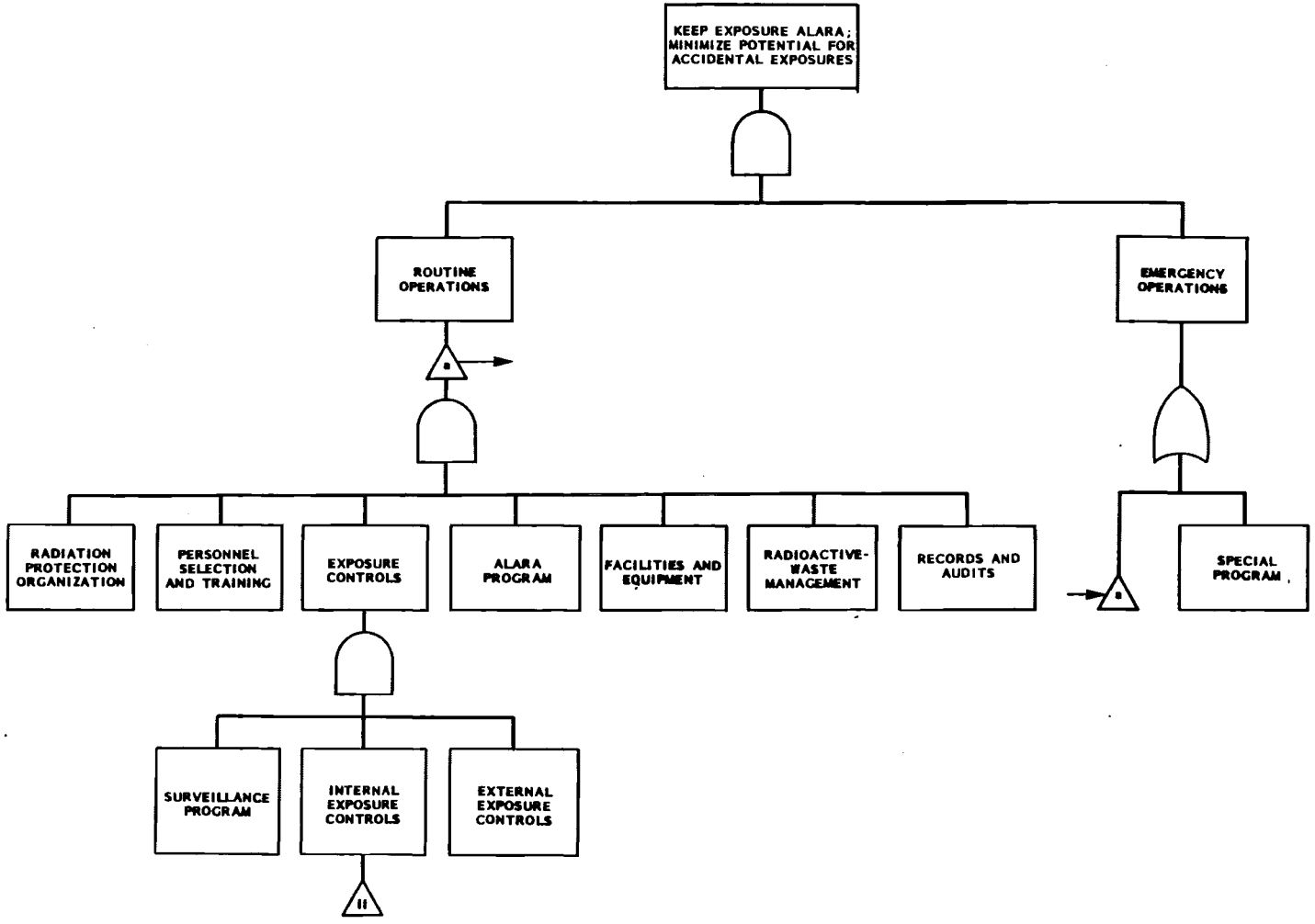


FIGURE 15.7. Upper Tiers for Logic Tree Depicting Radiation Protection Program

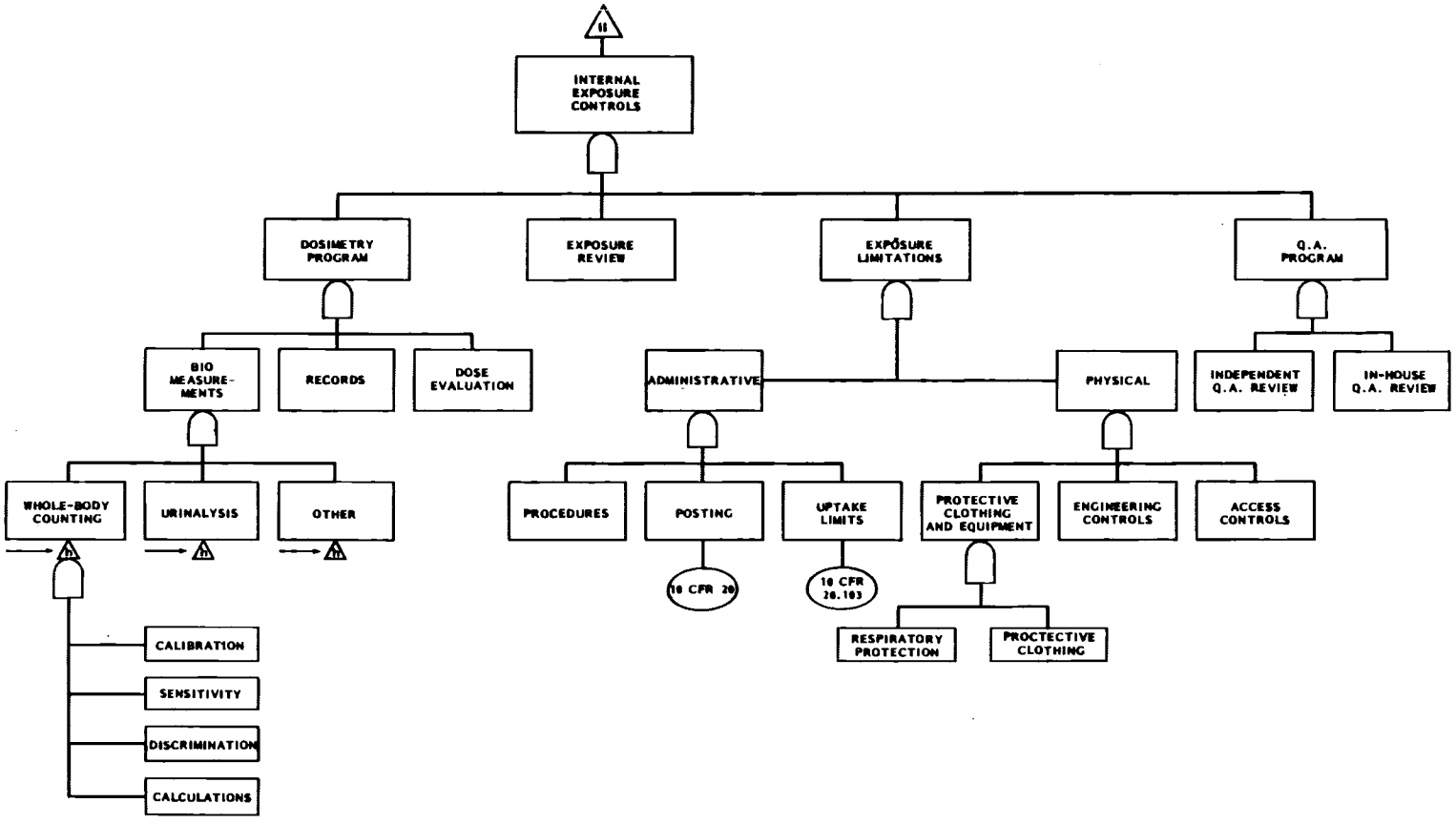


FIGURE 15.8. Logic Tree Tiers Depicting Internal-Exposure Control Program

18 trees, two of which interfaced with each of the remaining 16. The interfaces are usually designated by transfer functions (triangles with arrows and a letter or number) that indicate when data from one area should be used in the evaluation of another.

The analytical trees should be designed so that they graphically depict the total radiation protection program. To help the appraisers properly evaluate each area included in the trees, a checklist of questions, such as those in the previous section, is designed to accompany each element in every tier. The questions define the scope of the appraisal and ensure consideration of the essential elements of a radiation protection program. They are not intended to be an all-inclusive listing of the significant items for appraisal, but should provide the appraisers with the foundation upon which to evaluate the program. The appraisers should find that the answers to some questions lead them to a series of other questions that are not written in the appraisal guide.

The complexity of the appraisal process requires that the appraisers be familiar with a large number of regulations, regulatory guides, and industry standards. These documents will be useful in judging the adequacy of all or part of a specific area (e.g., dosimetry). In addition, the criteria used for designing the logic trees and for evaluating the program should be taken from DA and NRC rules and regulations, ANSI standards, National Council on Radiation Protection and Measurements (NCRP) guides, and recommendations of the ICRP and the International Commission on Radiation Units and Measurements (ICRU). However, the use of the logic tree system does not eliminate the need for professional judgment where standards and regulations do not provide sufficient detail; rather, its purpose is to help the appraisers clarify where their judgment is needed.

REFERENCES

American National Standards Institute (ANSI). 1966. Practice for Occupational Radiation Exposure Records Systems. ANSI N13.6, New York.

American National Standards Institute (ANSI). 1978. American National Standard for Internal Dosimetry for Mixed Fission and Activation Products. ANSI N343, New York.

American National Standards Institute (ANSI). 1978. Radiation Protection Instrumentation Test and Calibration. ANSI N323, New York.

American National Standards Institute (ANSI). 1980. American National Standard Criteria for Testing Personnel Dosimeter Performance. ANSI N13.11, New York.

American National Standards Institute (ANSI). 1980. Practices for Respiratory Protection. ANSI Z88.2, New York.

International Commission on Radiological Protection (ICRP). 1959. Report of Committee II on Permissible Dose for Internal Radiation. ICRP 2, Pergamon Press, Oxford.

U.S. Code of Federal Regulations. 1982. Title 10, Part 19, "Notices, Instructions and Reports to Workers; Inspections." U.S. Government Printing Office, Washington, D.C.

U.S. Code of Federal Regulations. 1982. Title 10, Part 20, "Standards for Protection Against Radiation." U.S. Government Printing Office, Washington, D.C.

U.S. Department of the Army, Headquarters. Safety - Ionizing Radiation Protection (Licensing, Control, Transportation, Disposal, and Radiation Safety). AR 385-11, Washington, D.C.

U.S. Department of the Army and Defense Logistics Agency. Medical Services - Control and Recording Procedures for Exposure to Ionizing Radiation and Radioactive Materials. AR 40-14, DLAR 1000.28, Washington, D.C.

U.S. Nuclear Regulatory Commission (NRC). 1975. "Operating Philosophy for Maintaining Occupational Radiation Exposure As Low As Reasonably Achievable." Regulatory Guide 8.10, Washington, D.C.

U.S. Nuclear Regulatory Commission (NRC). 1976. Manual of Respiratory Protection Against Radioactive Materials. NUREG-0041, Washington, D.C.

CHAPTER 16. REFERENCE DATA

Section 16.1 LIST OF ELEMENTS

Atomic Number	Symbol	Name	Atomic Number	Symbol	Name
1	H	hydrogen	53	I	iodine
2	He	helium	54	Xe	xenon
3	Li	lithium	55	Cs	cesium
4	Be	beryllium	56	Ba	barium
5	B	boron	57	La	lanthanum
6	C	carbon	58	Ce	cerium
7	N	nitrogen	59	Pr	praseodymium
8	O	oxygen	60	Nd	neodymium
9	F	fluorine	61	Pm	promethium
10	Ne	neon	62	Sm	samarium
11	Na	sodium	63	Eu	europium
12	Mg	magnesium	64	Gd	gadolinium
13	Al	aluminum	65	Tb	terbium
14	Si	silicon	66	Dy	dysprosium
15	P	phosphorus	67	Ho	holmium
16	S	sulfur	68	Er	erbium
17	Cl	chlorine	69	Tm	thulium
18	Ar	argon	70	Yb	ytterbium
19	K	potassium	71	Lu	lutetium
20	Ca	calcium	72	Hf	hafnium
21	Sc	scandium	73	Ta	tantalum
22	Ti	titanium	74	W	tungsten
23	V	vanadium	75	Re	rhenium
24	Cr	chromium	76	Os	osmium
25	Mn	manganese	77	Ir	iridium
26	Fe	iron	78	Pt	platinum
27	Co	cobalt	79	Au	gold
28	Ni	nickel	80	Hg	mercury
29	Cu	copper	81	Tl	thallium
30	Zn	zinc	82	Pb	lead
31	Ga	gallium	83	Bi	bismuth
32	Ge	germanium	84	Po	polonium
33	As	arsenic	85	At	astatine
34	Se	selenium	86	Rn	radon
35	Br	bromine	87	Fr	francium
36	Kr	krypton	88	Ra	radium
37	Rb	rubidium	89	Ac	actinium
38	Sr	strontium	90	Th	thorium
39	Y	yttrium	91	Pa	protactinium
40	Zr	zirconium	92	U	uranium
41	Nb	niobium	93	Np	neptunium
42	Mo	molybdenum	94	Pu	plutonium
43	Tc	technetium	95	Am	americium
44	Ru	ruthenium	96	Cm	curium
45	Rh	rhodium	97	Bk	berkelium
46	Pd	palladium	98	Cf	californium
47	Ag	silver	99	Es	einsteinium
48	Cd	cadmium	100	Fm	fermium
49	In	indium	101	Md	mendelevium
50	Sn	tin	102	No	nobelium
51	Sb	antimony	103	Lw	lawrencium
52	Te	tellurium			

Section 16.2 GREEK ALPHABET

Name	Upper Case	Lower Case
Alpha	A	α
Beta	B	β
Gamma	Γ	γ
Delta	Δ	δ
Epsilon	E	ε
Zeta	Z	ζ
Eta	H	η
Theta	Θ	θ
Iota	I	ι
Kappa	K	κ
Lambda	Λ	λ
Mu	M	μ
Nu	N	ν
Xi	Ξ	ξ
Omicron	O	o
Pi	Π	π
Rho	P	ρ
Sigma	Σ	σ
Tau	T	τ
Upsilon	Υ	υ
Phi	Φ	ϕ
Chi	Χ	×
Psi	Ψ	Ψ
Omega	Ω	ω

Section 16.3 ACRONYMS

AEC	U.S. Atomic Energy Commission
ALARA	as low as is reasonably achievable
ALI	annual limit of intake
AMC	Army Material Command
ANSI	American National Standards Institute
CAM	continuous air monitor
CFR	U.S. Code of Federal Regulations
CP	Cutie Pie
DA	U.S. Department of the Army
DAC	derived air concentration
DCP	disaster control plan
DD	U.S. Department of Defense
DF	decontamination factor
DOP	dioctyl-phthalate
DOT	U.S. Department of Transportation
DSA	Defense Supply Agency
DU	depleted uranium
EAL	emergency action level
ECC	emergency control center
ED	emergency director
EDTA	ethylene-diamine-tetra-acetic acid
EPA	U.S. Environmental Protection Agency
EPZ	emergency planning zone
GM	Geiger-Mueller
HEPA	high-efficiency particulate air
HEW	U.S. Department of Health, Education, and Welfare
HHS	Health and Human Services
HQ	Headquarter
HVL	half-value layer
IAEA	International Atomic Energy Agency
IATA	International Air Transport Association
ICRP	International Commission on Radiological Protection
ICRU	International Commission on Radiation Units and Measurements
IMCO	Inter-Governmental Maritime Consultative Organization
IRCC	Ionizing Radiation Control Committee
LET	linear energy transfer
LSA	low specific activity

MPBB	maximum permissible body burden
MPC	maximum permissible concentration
MSHA	Mine Safety and Health Administration
NBS	U.S. National Bureau of Standards
NCRP	National Council on Radiation Protection and Measurements
NIOSH	National Institute of Occupational Safety and Health
NRC	U.S. Nuclear Regulatory Commission
NTA	nuclear track emulsion
NTIS	National Technical Information Service
OSHA	Occupational Safety and Health Administration
PF	protection factor
RAC	radiological assessment and control
RAM	remote area monitor
RBE	relative biological effectiveness
RPO	Radiation Protection Officer
RSR	radioactive shipment record
SEE	specific effective energy
SI	international system of measurement units
SOP	standing operating procedure
STP	standard temperature and pressure (0°C, 760 mm Hg)
TI	transport index
TL	thermoluminescence
TLD	thermoluminescence dosimeter
TVL	tenth-value layer
WB	whole body

Section 16.4 ABBREVIATIONS AND SYMBOLS

ABBREVIATIONS

A	mass number
A	radionuclide activity or source activity
AF(T←S)	absorbed fraction of emitted energy, target from source
bis-MSB	p-bis-(O-methylstyryl) benzene
Bq	becquerel
Butyl-PBD	[2-(4'-tert-butylphenyl), 5-(4"-biphenylyl) - 1,3,4-oxadiazole]
C	Celsius
C	centigrade
c	centi- (10^{-2})
cc	cubic centimeter
Ci	Curie
cm_2	centimeter
cm^2	square centimeter
cpm_{-1}	counts per minute
cm^{-1}	reciprocal centimeter or 1/cm
D	radiation dose or absorbed radiation dose
$\dot{D}$	radiation dose rate
d	day
dis	disintegration
dpm	disintegrations per minute
dps	disintegrations per second
dx	differential of x
E	radiation energy
e	base of natural logarithms (2.71828)
e.g.	exempli gratia (for example)
esu	electrostatic unit
eV	electron volt
F	Fahrenheit
f_2	fraction of body burden in a given organ
ft	foot
g	gram
gal	gallon
Gy	gray
H	dose equivalent
H_T	committed dose equivalent to a target organ
$\dot{H}_T$	dose-equivalent rate to a target organ
h	Planck's constant
hr	hour
Hz	hertz

I	photon flux
I	radiation intensity
I_o	original radiation intensity
i.d.	inside diameter
i.e.	id est (that is)
in.	inch
J	joule
k	kilo- (10^3)
kg	kilogram
keV	kiloelectron volt
kV	kilovolt
L	liter
lb	pound
ln	natural logarithm
M	mega- (10^6)
m	mass
m	meter
m_2	milli- (10^{-6})
m^2	square meter
m^3	cubic meter
max	maximum
mCi	millicurie
MeV	million electron volt
min	minute
ml	milliliter
mm	millimeter
mR	milliroentgen
mrad	millirad
mrem	millirem
N	neutron number
N	number of radioactive atoms present at a time t
N	product of modifying factors
N_o	number of radioactive atoms originally present
n^o	any number
o.d.	outside diameter
oz	ounce
P	specific ionization, or number of ion pairs produced by radiation per path length
p	pico- (10^{-12})
pCi	picocurie
PPO	2,5-diphenyloxazole
POPOP	1,4-bis-[2-(5-phenyloxazolyl)]-benzene

Q	quality factor
q(t)	body burden at time t
R	roentgen
r	radius of a circle
S	source
S	surface area
s	distance
s	thickness
sec	second
SEE(T←S)	specific effective energy per disintegration, target from source
Sv	sievert
T	kinetic energy
T	target
t	time
$t_{1/2}$	radionuclide half-life
U	number of transformations in a source organ
u	mass unit
V	volt
v	velocity
wk	week
X	exposure
$\dot{X}$	exposure rate
x	times (multiplication)
Y	radiation yield
yr	year
Z	atomic number

GREEK SYMBOLS

α	alpha particle
β	beta particle
γ	gamma ray
Γ	gamma-ray constant
ε	effective absorbed energy per disintegration
ε	energy imparted by ionizing radiation
Θ	angle
λ	effective decay constant
λ	wavelength
λ_p	radionuclide decay constant

μ	linear attenuation coefficient
μ	micro- (10^{-6})
μ_{en}	mass energy absorption coefficient
μ_m	mass attenuation coefficient
μCi	microcurie
μm	micrometer
ν	frequency
ν	neutrino
π	pi (3.1416)
ρ	density
Σ	summation of

MATHEMATICAL SYMBOLS

$^\circ$	degree
%	percent
$\propto$	proportional to
$\cdot$	times (multiplication)

Section 16.5 SELECTED CONVERSIONS

MULTIPLY	BY	TO OBTAIN
Length		
centimeters	0.3937	inches
	3.28×10^{-2}	feet
feet	30.48	centimeters
	0.3048	meters
inches	2.54	centimeters
meters	3.281	feet
	39.37	inches
miles	5280	feet
Area		
barns	10^{-24}	square centimeters
square centimeters	10^{24}	barns
	1.076×10^{-3}	square feet
	0.155	square inches
square feet	929	square centimeters
	144	square inches
	9.29×10^{-2}	square meters
square inches	6.452	square centimeters
	6.944×10^{-3}	square feet
	6.452×10^{-4}	square meters
square meters	10.76	square feet
Volume		
cubic centimeters	6.102×10^{-2}	cubic inches
	3.531×10^{-5}	cubic feet
	2.642×10^{-4}	U.S. gallons
	10^{-3}	liters
cubic feet	2.832×10^{-2}	cubic meters
	7.481	U.S. gallons
	28.32	liters
cubic inches	16.39	cubic centimeters
	5.787×10^{-4}	cubic feet
	1.639×10^{-2}	liters
	4.329×10^{-3}	U.S. gallons
cubic meters	35.31	cubic feet
	2.642×10^{2}	U.S. gallons

MULTIPLY	BY	TO OBTAIN
Volume (cont'd)		
gallons, U.S.	231	cubic inches
	0.1337	cubic feet
	3.785×10^{3}	cubic centimeters
	3.785	liters
liters	3.53×10^{-2}	cubic feet
	61.02	cubic inches
	0.2642	U.S. gallons
	10^{3}	cubic centimeters
Mass		
grams	2.205×10^{-3}	pounds
kilograms	2.205	pounds
ounces	28.35	grams
	6.25×10^{-2}	pounds
pounds	453.6	grams
Energy		
British thermal units	1.055×10^{3}	joules
	0.252	kilogram-calories
electron volts	1.6×10^{-12}	ergs
	1.6×10^{-19}	joules
ergs	10^{-7}	joules
	6.24×10^{11}	electron volts
	6.24×10^{5}	megaelectron volts
gram-calories	3.968×10^{-3}	British thermal units
joules	10^{7}	ergs
	9.48×10^{-4}	British thermal units
kilogram-calories	3.968	British thermal units
megaelectron volts	1.6×10^{-6}	ergs
Radiation		
curies	3.7×10^{10}	becquerels
	2.22×10^{12}	disintegrations/minute
	3.7×10	disintegrations/second
	10^{3}	millicuries
	10^{6}	microcuries
	10^{-3}	kilocuries
becquerels	1	disintegrations/second
	2.7×10^{-11}	curies
disintegrations/minute	4.55×10^{-10}	millicuries
	4.55×10^{-7}	microcuries
disintegrations/second	2.7×10^{-8}	millicuries
	2.7×10^{-5}	microcuries
	1	becquerels

MULTIPLY	BY	TO OBTAIN
Radiation (cont'd)		
gray	10^2	rad
	1	joules/kilogram
microcuries	3.7×10^4	disintegrations/second
	2.22×10^6	disintegrations/minute
milliciuries	3.7×10^7	disintegrations/second
	2.22×10^4	disintegrations/minute
rad	10^{-2}	gray
	10^{-2}	joules/kilogram
	10^2	ergs/gram
rem	10^{-2}	sievert
	10^{-2}	joules/kilogram
roentgen	2.58×10^{-4}	coulombs/kilogram
	1	electrostatic units/ cubic centimeter air (at STP)
	2.082×10^9	ion pairs/cubic centimeter air (at STP)
	1.61×10^{12}	ion pairs/gram air
	7.03×10^4	MeV/cubic centimeter air (at STP)(a)
	5.44×10^7	MeV/gram air(a)
	87_2	ergs/gram air(a)
sievert	10^2	rem
	1	joules/kilogram
Temperature		
degrees Celsius	1.8	degrees Fahrenheit - 32
degrees Fahrenheit - 32	0.5555	degrees Celsius

(a) Assuming that the average energy expended per ion pair formed is 5.4×10^{-11} ergs (34 eV).

Section 16.6 FREQUENTLY USED CONSTANTS

Avogadro's number	$N = 6.0220 \times 10^{23}\ mol^{-1}$
Velocity of light	$c = 2.997925 \times 10^{8}\ m/sec$
Electronic charge	$e = 0.16022 \times 10^{-18}\ C$
Planck's constant	$h = 6.626 \times 10^{-34}\ J \cdot sec$
	$= 6.626 \times 10^{-27}\ erg \cdot sec$
	$= 0.41355 \times 10^{-14}\ eV \cdot sec$
Mass of electron	$m_e = 0.910953 \times 10^{-30}\ kg$
Mass of proton	$m_p = 0.167265 \times 10^{-26}\ kg$
Mass of neutron	$m_n = 0.167495 \times 10^{-26}\ kg$

Section 16.7 ADDRESSES FOR ORDERING REFERENCE DOCUMENTS

American National Standards Institute (ANSI)
Sales Department
American National Standards Institute
1430 Broadway
New York, NY 10018

Code of Federal Regulations (CFR)
Superintendent of Documents
U.S. Government Printing Office
Washington, DC 20402

International Atomic Energy Agency (IAEA)
UNIPUB
345 Park Avenue South
New York, NY 10010

International Commission on Radiation Units and Measurements (ICRU)
ICRU Publications
P.O. Box 30165
Washington, DC 20014

International Commission on Radiological Protection (ICRP)
Pergamon Press
Maxwell House
Fairview Park
Elmsford, NY 10523

National Council on Radiation Protection and Measurements (NCRP)
NCRP Publications
P.O. Box 30175
Washington, DC 20014

National Technical Information Service (NTIS)
U. S. Department of Commerce
5285 Port Royal Road
Springfield, VA 22151

U.S. Department of Transportation (DOT)
Superintendent of Documents
U.S. Printing Office
Washington, DC 20402

U.S. Nuclear Regulatory Commission (NRC)
Superintendent of Documents
U.S. Printing Office
Washington, DC 20402

Section 16.8 GLOSSARY

ABSORPTION:	The process by which radiation imparts some or all of its energy to any material through which it passes.
ACCELERATOR (PARTICLE ACCELERATOR):	A device for imparting large quantities of kinetic energy to electrically charged particles such as electrons, protons, and helium ions.
ACTIVATION:	The process of inducing radioactivity by irradiation.
ACTIVITY:	The number of nuclear transformations occurring in a given quantity of material per unit time. The unit of measure is the curie (Ci).
ACUTE EXPOSURE:	Radiation exposure of short duration.
AGREEMENT STATE:	Any state in the United States with which NRC has made an effective agreement under Subsection 274(b) of the Atomic Energy Act of 1954, as amended, relative to the licensing and control of radioactive material used or produced within that state.
AIRBORNE CONTAMINATION:	The term applied to radioactive contamination loose in the air, filtered from the air, or deposited from the air, as contrasted with contamination spread by splashing, dripping, etc.
AIR-WALL IONIZATION CHAMBER:	An ionization chamber in which the materials of the wall and electrodes are so selected as to produce ionization essentially equivalent to that in a free-air ionization chamber. This is possible only over limited ranges of photon energies. Such a chamber is more appropriately termed an "air-equivalent ionization chamber."
ALARA:	An acronym for "as low as is reasonably achievable"; refers to an operating philosophy in which occupational exposures are reduced as far below specified limits as is reasonably achievable.
ALPHA PARTICLE:	A charged particle that is emitted from the nucleus of an atom and that has a mass and charge equal in magnitude to those of a helium nucleus, i.e., two protons and two neutrons.

AMPLIFICATION: As related to radiation detection instruments, the process (gas, electronic, or both) by which ionization effects are magnified to a degree suitable for their measurement.

ANALYZER, PULSE HEIGHT: An electronic circuit that sorts and records pulses according to their height.

ANGULAR DEPENDENCE: The varying ability of an instrument to accurately measure radiation, depending on its orientation with respect to the radiation field.

ANODE: A positive electrode; the electrode to which negative ions are attracted.

APPRAISAL: A comprehensive evaluation of the overall adequacy and effectiveness of a radiation protection program.

ARTIFICAL RADIOACTIVITY: Manmade radioactivity produced by particle bombardment or electromagnetic irradiation, as opposed to natural radioactivity.

ATOM: The smallest unit of an element that is capable of entering into a chemical reaction.

ATOMIC NUMBER: The number of protons in the nucleus of a neutral atom of a nuclide.

ATTENUATION: The process by which a beam of radiation is reduced in intensity or energy when passing through some material.

AUTHORIZED MATERIAL: Radioactive material not requiring a specific NRC license. The receipt, possession, use, or transfer of radioactive material requires specific authorization or permit by a specific agency or service organization.

AVALANCHE: The multiplicative process in which a single charged particle accelerated by a strong electric field produces additional charged particles through collision with neutral gas molecules. This cumulative increase of ions is also known as "Townsend ionization" or "Townsend avalanche."

BACKGROUND RADIATION: Radiation arising from radioactive material other than the one directly under consideration. Background radiation due to cosmic rays

	and natural radioactivity is always present. There may also be background radiation due to the presence of radioactive substances in other parts of a building, in the building material itself, etc.
BEAM:	A unidirectional or approximately unidirectional flow of electromagnetic radiation or of particles.
BECQUEREL:	The SI unit of activity equal to a nuclear disintegration rate of 1 disintegration per second.
BETA PARTICLE:	A charged particle emitted from the nucleus of an atom, with a mass and charge equal in magnitude to those of the electron.
BIOASSAY:	An evaluation of the amount of radioactivity taken into the body.
BREMSSTRAHLUNG:	Secondary photon radiation produced by the deceleration of charged particles passing through matter.
BYPRODUCT MATERIAL:	Any material (except special nuclear material) made radioactive by either exposure to radiation, or the process of producing or using special nuclear material.
CALIBRATION:	The determination of a measuring instrument's variation from a standard, to ascertain necessary correction factors.
CATHODE:	A negative electrode; the electrode to which positive ions are attracted.
CELL (BIOLOGICAL):	The fundamental unit of structure and function in organisms.
CHAIN REACTION):	Any chemical or nuclear process in which some products or energy released by the process are instrumental in the continuation or magnification of the process.

CHARACTERISTICS (DISCRETE) RADIATION: Radiation originating from an atom after the removal of an electron or the excitation of the nucleus. The wavelength of the emitted radiation is specific, depending only on the nuclide and the particular energy levels involved.

CHRONIC EXPOSURE: Radiation exposure of long but not necessarily continuous duration.

COLLECTIVE DOSE EQUIVALENT: The sum of dose equivalents received by a given population or group of workers, expressed in units of person-rem.

COLLISION: An encounter between two subatomic particles (including photons) that changes the initial momentum and energy conditions. The products of the collision need not be the same as the initial systems.

COMMODITY (RADIOACTIVE): An item of government property made up in whole or in part of radioactive materials. A national stock number (NSN) (formerly called a federal stock number (FSN)) or part number is assigned to items that contain radioactive material in excess of 0.01 μCi.

COMPOUND: A distinct substance formed by the union of two or more ingredients in definite proportions by weight.

COMPTON EFFECT: An attenuation process observed for x or gamma radiation in which an incident photon interacts with an orbital electron of an atom to produce a recoil electron and a scattered photon with an energy less than that of the incident photon.

CONDENSER R-METER: An instrument consisting of an air-wall ionization chamber together with auxiliary equipment for charging and measuring its voltage. It is used as an integrating instrument for measuring the exposure of x or gamma radiation in roentgens (R).

CONTAMINATION (RADIOACTIVE): The deposition of radioactive material in any place where it is not desired, and particularly in any place where its presence might be harmful.

COUNT (RADIATION MEASUREMENTS):	The external indication of a device designed to enumerate ionizing events. It may refer to a single detected event or to the total number registered in a given period of time. The term is often used erroneously to designate a disintegration, ionizing event, or voltage pulse.
COUNTER:	A gas-filled radiation detector (chamber or tube) connected to an auxiliary electronic circuit in such a way that individual pulses from ionization events inside the chamber register in an external counting device.
CRITICAL:	Capable of sustaining (at a constant level) a chain reaction. "Prompt critical" means sustaining a chain reaction without the aid of delayed neutrons.
CRITICAL ORGAN:	The organ of the body receiving a specified radioisotope that results in the greatest physiological damage to the body. For exposure to ionizing radiation from external sources, the critical organs are the skin, blood-forming organs, gonads, and eyes.
CROSS-CONTAMINATION:	Contamination not from an original source, but acquired from another contaminated object. The term is used in laboratory, bioassay, and counting-room work to refer to the spread of contamination from contaminated samples to relatively uncontaminated samples, thus giving erroneously high readings to the latter.
CUMULATIVE DOSE (RADIATION):	The total dose resulting from repeated exposures to radiation.
CURIE:	The special unit of activity (abbreviated Ci). One curie equals exactly 3.7×10^{10} nuclear disintegrations per second.
DAUGHTER:	Synonym for decay product.
DECAY CONSTANT:	The fraction of the number of atoms of a radioactive nuclide that decay per unit time.
DECAY, RADIOACTIVE:	The disintegration of the nucleus of an unstable nuclide by the spontaneous emission of charged particles and/or photons.
DECONTAMINATION:	The reduction or removal of radioactive contamination from any given surface.

DELTA RAY: Any secondary ionizing particle ejected by recoil when a primary ionizing particle passes through matter.

DETECTOR, Ge(Li): A solid-state detector in which the crystal used is germanium (Ge) with a minute quantity of lithium (Li) impurity added to stabilize the action. (It is sometimes referred to as a "jelly" detector.)

DETECTOR, INTEGRATING: A detector that measures a total accumulated radiation quantity (such as exposure or dose) rather than the rate of accumulation of the radiation. Devices that accumulate and hold charges (e.g., electrometers) and that indicate measures proportional to the total dose are of this type. Examples of integrating detectors are electrometers, film badges, pocket dosimeters, and neutron activation detectors.

DETECTOR, RADIATION: Any device for converting radiant energy to a form more suitable for observation. An instrument used to determine the presence, and sometimes the amount, of radiation.

DETECTOR, SCINTILLATION: A radiation detector whose response is a light signal generated by the incident radiation and a scintillating medium. The light signal is transformed into an electronic signal through an adjacent, optically coupled, photo-sensitive device such as a photomultiplier tube.

DETECTOR, SOLID-STATE: A generic name for a radiation detector that uses solid-state devices, such as the semiconductors germanium or silicon, which respond to incident radiation with an electronically measurable pulse.

DETECTOR, TRACK (ETCH): A device that records the paths of heavy charged particles in a transparent solid. The tracks may be directly visible, or they may be enhanced by etching with an appropriate reagent (such as potassium hydroxide for etching cellulose acetate).

DISINTEGRATION, NUCLEAR: A spontaneous nuclear transformation (radioactivity) characterized by the emission of energy and/or mass from the nucleus. When numbers of nuclei are involved, the process is characterized by a definite half-life.

DOSE: A general term denoting the quantity of radiation or energy absorbed. For special purposes, the term must be appropriately qualified. If unqualified, it refers to absorbed dose.

DOSE, ABSORBED: The amount of energy imparted to matter in a volume element by ionizing radiation, divided by the mass of irradiated material in that element. Also called dose. The common unit of absorbed dose is the rad, which is equal to 100 ergs of absorbed energy per gram of material (or 0.01 J/kg). The SI unit of absorbed dose is the gray, which is equal to 100 rad or to 1 joule of absorbed energy per kilogram of material.

DOSE, WHOLE-BODY: The average uniform absorbed dose or dose equivalent received by a person whose whole body is exposed to ionizing radiation from an external source.

DOSE EQUIVALENT: The product of the absorbed dose, the quality factor, and other modifying factors necessary to evaluate the effects of irradiation received by exposed persons. This unit of measure takes into account the particular characteristics of the exposure. The common unit of dose equivalent is the rem. The SI unit is the sievert. Absorbed doses of different types of radiation are not additive, but dose equivalents are, because they express on a common scale the amount of damage incurred.

DOSE METER, INTEGRATING: An ionization chamber and measuring system designed to determine the total radiation administered during an exposure. In medical radiology, the chamber is usually designed to be placed on the patient's skin. A device may be included to terminate the exposure when it has reached a particular value.

DOSIMETER: An instrument to detect and measure accumulated radiation exposure. In common usage, a pencil-sized ionization chamber with a self-reading electrometer, used for personnel monitoring.

DOSIMETER, PERSONAL: A dosimeter of small size carried by a person to determine the exposure, absorbed dose, and/or dose equivalent received during the carrying time. Also called personal exposure meter.

DOSIMETER, POCKET: A dosimeter the shape and size of a fountain pen with a clip, to be worn in the pocket like a fountain pen.

DOSIMETER, THERMOLUMINESCENCE: An integrating detector that utilizes a phosphor sensitive to ionizing radiation. The phosphor stores the energy of the ionization within itself and releases it as low-energy photons (light) when heated. The total amount of light released is proportional to the total absorbed dose.

DOSIMETRY, PHOTOGRAPHIC: The determination of cumulative radiation dose using photographic film and density measurement.

EFFICIENCY (OF COUNTERS): A measure of the probability that a count will be recorded when radiation is incident on a detector. Uses of this term vary considerably, so it is well to ascertain which factors (window transmission, sensitive volume, energy dependence, etc.) are included in a given case.

ELASTIC COLLISION: A collision in which there is no change either in the internal energy of each participating system or in the sum of their kinetic energies of translation.

ELECTRODE: A conductor used to establish electrical contact with a nonmetallic part of a circuit.

ELECTRON: A stable elementary particle that has an electric charge equal to $\pm 1.60210 \times 10^{-19}$ coulomb and a rest mass equal to 98.1091×10^{-31} kg.

ELECTRON VOLT: A unit of energy equivalent to the energy gained by an electron in passing through a potential difference of 1 volt. Larger multiple units of the electron volt are frequently used: keV for thousand or kilo-electron volts; MeV for million or mega-electron volts. 1 eV = 1.6×10^{-12} erg.

ELEMENT: A category of atoms all of which have the same atomic number.

EMULSION, NUCLEAR: A photographic emulsion specially designed to permit observation of the individual tracks of ionizing particles.

ENERGY DEPENDENCE: The characteristic response of a radiation detector to a given range of radiation energies or wavelengths, compared with the response of a standard free-air chamber.

ENRICHED MATERIAL: (1) Material in which the relative amount of one or more isotopes of a constituent has been increased.

(2) Uranium in which the abundance of the ^{235}U isotope is increased above normal.

EXCITED STATE (OF A NUCLEUS): An unstable condition of the nucleus of an atom after the entrance of a nuclear particle or gamma-ray photon.

EXPOSURE: (1) The incidence of radiation upon inanimate or living matter by intent or accident.

(2) For x or gamma radiation, the sum of the electrical charges of all the ions of one sign produced in air when all electrons liberated by photons in a suitable small volume of air are completely stopped in air, divided by the mass of air in the volume.

The unit of exposure is the roentgen (R).

EXPOSURE RATE: (1) The exposure divided by the time over which it was accumulated.

(2) The increment of exposure during a suitably small interval of time, divided by that interval of time.

The usual unit of exposure rate is roentgens per hour (R/hr).

EXTERNAL RADIATION: Radiation from a source outside the body.

FALLOUT: Radioactive debris from a nuclear detonation, which is airborne or has been deposited on the earth. Special forms of fallout are "dry fallout," "rainout," and "snowout."

FILTER (RADIOLOGY): Primary--A sheet of material, usually metal, placed in a beam of radiation to absorb preferentially the less penetrating components.

Secondary--A sheet of material of low atomic number (relative to the primary filter) placed in the filtered beam of radiation to remove characteristic radiation produced by the primary filter.

FINGER DOSIMETER: A dosimeter in the form of a ring to be worn by personnel to determine radiation doses to the hands.

FISSILE: A nuclide capable of undergoing fission by interaction with slow neutrons.

FISSILE MATERIAL: Plutonium-238, plutonium-239, plutonium-241, uranium-233, uranium-235, or any material containing any of the foregoing [49 CFR 173.389(a) and 173.398(a)].

FISSION (NUCLEAR): A nuclear transformation characterized by splitting of a nucleus into at least two other nuclei and the release of a relatively large amount of energy.

FISSIONABLE: Pertaining to a nuclide that is capable of undergoing fission by any process.

FISSION PRODUCTS: Elements or compounds resulting from fission.

FLUENCE: The number of particles passing through a unit cross-sectional area.

FLUORESCENCE: The emission of radiation of particular wavelengths by a substance as a result of the absorption of radiation of shorter wavelengths. This emission occurs essentially only during the irradiation.

FLUOROSCOPE: A fluorescent screen, suitably mounted with respect to an x-ray tube for ease of observation and protection, used for indirect visualization (by x rays) of internal organs in the body or internal structures in apparatus or in masses of material.

GAS AMPLIFICATION: As applied to gas-ionization instruments for detecting radiation, the ratio of the charge collected to the charge produced by the initial ionizing event.

GEIGER-MUELLER COUNTER: A highly sensitive, gas-filled radiation-measuring device. It operates at voltages high enough to produce avalanche ionization.

GEOMETRY, GOOD: In nuclear physics measurements, an arrangement of source and detecting equipment that introduces little error when a finite source size and finite detector aperture are used.

GEOMETRY, POOR: In a nuclear experiment, an arrangement in which the angular aperture between the source and detector is large, introducing into the measurement a comparatively large uncertainty for which a correction may be necessary.

GEOMETRY (RADIATION): A nuclear physics term referring to the physical relationship and symmetry of the parts of a radiation detection assembly. Counting efficiency is closely related to geometry.

GLOW CURVE: In thermoluminescence dosimetry, a graph of the released luminescence photon fluence as a function of temperature or time of heating. The area under the bell-shaped curve plotted against time is proportional to the total absorbed dose or exposure.

GLOW PEAK: In thermoluminescence dosimetry, the time or temperature during heating of a thermoluminescence phosphor at which the release rate of the luminescence photons is maximum.

GRAY: The SI unit of absorbed dose, equal to the absorbed energy from ionizing radiation of 1 joule/kg, and equal to 100 rads.

GROUND STATE: The state of a nucleus, atom, or molecule at its lowest energy. All other states are "excited."

HALF-LIFE, BIOLOGICAL: The time required for the body to eliminate one-half of an administered dosage of any substance by processes of elimination. Approximately the same for both stable and radioactive isotopes of a particular element.

HALF-LIFE, EFFECTIVE: The time required for a radioactive element in an animal body to be diminished 50% as a result of the combined action of radioactive decay and biological elimination.

Effective half-life

$$= \frac{\text{Biological half-life} \times \text{Radioactive half-life}}{\text{Biological half-life} + \text{Radioactive half-life}}$$

HALF-LIFE, RADIOACTIVE: The time required for a radioactive substance to lose 50% of its activity by decay. Each radionuclide has a unique half-life.

HALF-VALUE LAYER (HALF THICKNESS) (HVL): The thickness of a specified substance that, when introduced into the path of a given beam of radiation, reduces the exposure rate by one-half.

HEALTH PHYSICS: A science and profession devoted to protecting man and the environment against unnecessary radiation exposure.

HOLE (SOLID-STATE THEORY): A position in the valence bands of semiconductor or insulating materials denoting the absence of an electron. Such a position carries a positive charge that (like an electron) is able to migrate within the band.

INDUCED RADIOACTIVITY: Radioactivity produced in a substance after bombardment with neutrons or other particles. The resulting activity is "natural radioactivity" if formed by nuclear reactions occurring in nature, and "artificial radioactivity" if the reactions are caused by man.

INELASTIC COLLISION: A collision in which there are changes both in the internal energy of one or more of the colliding systems and in the sums of the kinetic energies of translation before and after the collision.

INFRARED RADIATION: Invisible thermal radiation whose wavelength is longer than the red segment of the visible spectrum.

INGESTION (OF RADIOACTIVITY): The entry of radioactivity into the body through the mouth.

INHALATION (OF RADIOACTIVITY): The entry of radioactivity into the body through the breathing of airborne radioactive particulate matter.

INTENSITY: The amount of energy per unit time passing through a unit area perpendicular to the line of propagation at the point in question.

INTENSITY, RADIATION: A generic term for the magnitude of a radiation quantity.

INTENSITY, SOURCE:	A generic term for the magnitude of a source emission rate. The source intensity of a radioisotope source is related to its disintegration rate in curies or bequerels.
INTERNAL RADIATION:	Radiation from a source within the body (as a result of the deposition of radionuclides in body tissues).
IN-VIVO COUNTING:	Measurements of internal radiation made at the surface (outside) of the body and based on the fact that radioisotopes emit radiation that can traverse the tissues and be measured outside the organism. In-vivo counting is synonymous with whole-body counting.
ION:	An atomic particle or atom bearing an electric charge, either negative or positive.
IONIZATION:	The process by which a neutral atom or molecule acquires a positive or negative charge.
IONIZATION CHAMBER:	An instrument designed to measure a quantity of ionizing radiation in terms of the charge of electricity associated with ions produced within a defined volume.
IONIZING RADIATION CONTROL COMMITTEE:	A group of qualified personnel officially appointed by a commander to set local policy and to guide the radiation protection program.
IONIZING-RADIATION-PRODUCING DEVICES:	Electronic devices that are capable of making ionizing radiation. Examples are x-ray machines, linear accelerators, and electron microscopes.
ION PAIR:	Two particles of opposite charge, usually referring to the electron and the positive atomic or molecular residue resulting from the interaction of ionizing radiation with the orbital electrons of atoms.
ISOMERS:	Nuclides with the same number of neutrons and protons but capable of existing, for a measurable time, in different quantum states with different energies and radioactive properties. Commonly, the isomer of higher energy decays to one with lower energy by the process of isometric transition.

ISOTOPES:	Nuclides that have the same number of protons in their nuclei, hence the same atomic number, but that differ in the number of neutrons and therefore in the mass number. Isotopes of a particular element have almost identical chemical properties. The term should not be used as a synonym for nuclide.
JOULE:	The unit for work and energy, equal to 10^7 ergs.
LATENT PERIOD:	The interval of seeming inactivity between the time of irradiation and the appearance of an effect.
LEAKAGE RADIATION:	Radiation emerging from a surface, a body of material, or a region in space.
LICENSE (SPECIFIC):	A document issued by NRC under 10 CFR that gives the bearer the right to procure, receive, store, transfer, use, export, and import specified radioactive items under specific terms.
LICENSE-EXEMPT MATERIAL ITEMS:	Radioactive material not subject to NRC regulations, or exempt from NRC licensing under 10 CFR.
LICENSED MATERIAL:	Source, special nuclear, or byproduct material received, stored, possessed, used, or transferred under a general or specific license issued by NRC or an Agreement State.
LINEAR ACCELERATOR:	A device for accelerating charged particles. It employs alternate electrodes and gaps arranged in a straight line, so proportioned that when potentials are varied in the proper amplitude and frequency, particles passing through the waveguide receive successive increments of energy.
LINEAR ENERGY TRANSFER (LET):	The linear rate of loss of energy (locally absorbed) over distance by an ionizing particle moving in a material medium. The usual unit of LET is keV/μm.
MANIPULATOR:	Mechanical hands or some other device for performing work behind a barrier or in a glove box.
MAN-REM:	A unit of population dose equivalent or collective dose equivalent. The number of man-rems of dose equivalent is equal to the product of the population and the average dose equivalent in rem common to that population.

MAXIMUM CREDIBLE ACCIDENT: The worst accident in a reactor or nuclear energy installation that, by agreement, need be taken into account in devising protective measures.

MICROWAVE: An electromagnetic wave with a wavelength of approximately 1 millimeter to 1 meter and corresponding to frequencies of about 300 to 300,000 megacycles per second.

MOLECULE: The smallest unit of a compound, consisting of two or more atoms held together by chemical bonds.

MONITORING: Periodic or continuous determination of the amount of ionizing radiation or radioactive contamination present in an occupied region.

MONTE CARLO METHOD: A method permitting the computer solution of physics problems, such as those of neutron transport, by determining the history of a large number of elementary events by the application of the mathematical theory of random variables.

NATURAL RADIOACTIVITY: The property of radioactivity exhibited by more than 50 naturally occurring radionuclides.

NATURALLY OCCURRING RADIOACTIVE MATERIALS: Radioactive isotopes, such as radium and radon, that are found in nature but are not classified as source material.

NEUTRINO: A neutral particle of very small rest mass originally postulated to account for the continuous distribution of energy among particles in the beta-decay process.

NEUTRON: One of three elementary particles, which is part of all nuclei heavier than hydrogen.

NUCLEON: The common name for a constituent particle of the nucleus. Applied to a proton or neutron.

NUCLEUS (NUCLEAR): That part of an atom in which the total positive electric charge and most of the mass are concentrated.

NUCLIDE: A species of atom characterized by the constitution of its nucleus. The nuclear constitution is specified by the number of protons (Z), number of neutrons (N), and energy content; or, alternatively, by the atomic number (Z), mass number (A = N + Z), and atomic mass. To be

regarded as a distinct nuclide, the atom must be capable of existing for a measurable time. Thus, nuclear isomers are separate nuclides, whereas promptly decaying excited nuclear states and unstable intermediates in nuclear reactions are not so considered.

PAIR PRODUCTION: An absorption process for x and gamma radiation in which the incident photon is annihilated in the vicinity of the nucleus of the absorbing atom, with subsequent production of an electron and positron pair. This reaction occurs only for incident photon energies exceeding 1.02 MeV.

PARENT: A radionuclide which, upon disintegration, yields a specified nuclide, either directly or as a later member of a radioactive series.

PERSONNEL MONITOR: An instrument that measures a radiation quantity proportional to dose equivalent, for use by an individual working in a radiation area.

PHANTOM: A volume of material approximating as closely as possible the density and effective atomic number of body tissue. Ideally, a phantom should absorb radiation in the same way tissues does. Radiation dose measurements made within or on a phantom provide a means of determining the radiation dose within or on a body under similar exposure conditions. Some materials commonly used in phantoms are water, Masonite, pressed wood, and beeswax.

PHOSPHORESCENCE: The emission of radiation by a substance as a result of the previous absorption of radiation of shorter wavelength. In contrast to fluorescent emissions, the phosphorescent emissions may continue for a considerable time after cessation of the exciting irradiation.

PHOTOELECTRIC EFFECT: The process by which a photon ejects an electron from an atom. All the energy of the photon is absorbed in ejecting the electron and in imparting kinetic energy to it.

PHOTON: A quantity of electromagnetic energy (E) whose value in joules is the product of its frequency (O) in hertz and Planck's constant (h). The equation is $E = hO$.

PIG: A container, usually lead, used to ship or store radioactive materials.

PRIMARY IONIZATION: (1) In collision theory: the ionization produced by primary particles, as contrasted with total ionization, which includes the secondary ionization produced by delta rays.

(2) In counter tubes: the total ionization produced by incident radiation without gas amplification.

PROPORTIONAL COUNTER: A gas-filled radiation detector tube operated in that range of applied voltage in which the charge collected per isolated count is proportional to the charge liberated by the original ionizing event. The range of applied voltage depends upon the type and energy of the incident radiation.

PROTECTIVE CLOTHING: The clothing worn by radiation workers to prevent radioactive contamination of the body or personal clothing.

PROTECTIVE EQUIPMENT: Safety devices such as goggles or clothing used to do a job safely.

PROTON: An elementary nuclear particle with a positive electric charge equal numerically to the charge of the electron and a mass of 1.007277 mass units.

PURGING: The removal of material from a system or pipe by adding another material, such as blowing with air.

PYROPHORIC: Igniting spontaneously on exposure to air.

QUALITY FACTOR (Q): The factor dependent on linear energy transfer by which absorbed doses are multiplied to obtain (for radiation protection purposes) a quantity that expresses the effect of the absorbed dose on a common scale for all ionizing radiations.

QUENCHING: The process of inhibiting continuous or multiple discharge in a counter tube that uses gas amplification.

RAD: The unit of absorbed dose equal to 0.01 J/kg in any medium.

RADIATION: Energy traveling through space in the form of waves, particles, or bundles called photons.

RADIATION AREA: An area or item of equipment requiring access control for personnel protective purposes; an area or item of equipment presenting personnel hazards due to radiation or contamination.

RADIATION, DIRECT: Radiation reaching a given location directly from an emitting source without collision or energy degradation. Also called unscattered or uncollided radiation.

RADIATION, INDIRECT: Radiation reaching a given location after having been scattered at least once. Also called scattered radiation.

RADIATION, IONIZING: Radiation composed of particles that are themselves ionized (directly ionizing radiation) or that are able to ionize other atoms by reaction with them (indirectly ionizing radiation).

RADIATION, PRIMARY: (1) Radiation emitted by a primary nuclear reaction source (as opposed to radiation emitted by subsequent nuclear or atomic interactions as a result of primary radiation interactions).

(2) Radiation originating within an emitting source (such as the core of a nuclear reactor).

RADIATION, SCATTERED: Radiation reaching a given location after having undergone at least one scattering. See also radiation, indirect.

RADIATION, SECONDARY: Radiation emitted by some nuclear or atomic process as a result of previous nuclear or atomic interactions by a primary radiation source. Example: capture-gamma radiation.

RADIATION CONTROL OFFICER: An officer, enlisted person, or DA civilian employee appointed by each major Army commander to manage the radiation protection program for the major command.

RADIATION HAZARD: The presumed risk or deleterious effects attributable to deliberate, accidental, or natural exposure to radiation.

RADIATION PROTECTION OFFICER: A person appointed by the commander to give advice on the hazards of ionizing radiation and to supply effective ways to control these hazards.

RADIOACTIVE CONTROLLED ITEMS:	All commodities, components, and end items containing radioactive material that are controlled with respect to maintenance, disposal, and bulk storage. Items requiring additional controls are listed in 10 CFR 30.71.
RADIOACTIVE INDIVIDUALLY CONTROLLED ITEMS:	Items that are assigned national stock numbers and must be controlled to the extent that their integrity and location are known by the licensee or designated agent (control point) at all times.
RADIOACTIVE MATERIAL:	Any material or combination of materials that spontaneously gives off ionizing radiation. This includes natural elements such as radium, and accelerator-made radionuclides.
RADIOACTIVE MATERIAL CONTROL POINT:	Any Army element (including the RCO) that has been designated by a major Army commander to control radioactive items within the command.
RADIOACTIVE WASTE:	Waste materials that include the following: a. property contaminated to the extent that decontamination is economically unsound b. surplus radioactive material whose sale, transfer, or donation is prohibited c. surplus radioactive material that is determined to be unwanted after being advertised as surplus d. waste that is radioactive due to production, possession, or use of radioactive material.
RADIOACTIVITY:	A natural and spontaneous process by which the unstable atoms of an element emit or radiate excess energy from their nuclei as particles or photons and thus change (or decay) to atoms of a different element or to a lower energy form of the original element.
RADIOBIOLOGY:	The branch of biology that deals with the effects of radiation on biological systems.
RADIOCHEMISTRY:	The aspects of chemistry connected with radionuclides and their properties, with the behavior of minute quantities of radioactive materials, and with the use of radionuclides in the study of chemical problems.
RADIOGRAPH:	A shadow picture produced by passing x rays or gamma rays through an object and recording the variations in the intensity of the emergent rays on photographic or sensitized film.

RADIOLOGY: The branch of medicine that deals with the diagnostic and therapeutic applications of radiant energy, including x rays and radionuclides.

RADIOPHARMACEUTICAL: A pharmaceutical compound that has been tagged with a radionuclide.

RADIOSENSITIVITY: The relative susceptibility of cells, tissues, organs, organisms, or any living substance to the injurious action of radiation.

REM: A special unit of dose equivalent. The dose equivalent in rems is numerically equal to the absorbed dose in rads multiplied by the quality factor and any other necessary modifying factors.

RESTRICTED AREA: Any area in a radiation facility to which access is controlled by the licensee for purposes of protecting individuals from exposure to radiation and radioactive materials.

ROENTGEN: One roentgen is the quantity of charge liberated by x or gamma radiation and is equal to 2.58×10^{-4} coulombs per kilogram of dry air. It is equivalent to the energy absorption of x or gamma radiation of 87.7 ergs/g of air or 96.5 ergs/g of tissue (0.00877 J/kg and 0.00965 J/kg).

SATURATION (IONIZATION CHAMBER): The condition in an ionization chamber when the applied voltage is sufficient to collect all the ions formed from the absorption of radiation, but insufficient to cause ionization by collisions.

SCATTERING: Change of direction of subatomic particles or photons as a result of a collision or interaction.

SEALED SOURCE: Any radioactive material that is permanently bonded or fixed in a capsule or matrix designed to prevent the release or dispersal of the material under the most severe conditions encountered in normal use or handling.

SECONDARY IONIZATION: Ionization produced by delta rays.

SELF-ABSORPTION: The absorption of radiation (emitted by radioactive atoms) by the material in which the atoms are located; in particular, the absorption of radiation within a sample being assayed.

SERIES, RADIOACTIVE: A succession of nuclides, each of which transforms by radioactive disintegration into the next until a stable nuclide results. The first member is called the "parent," the intermediate members are called "daughters," and the final stable member is called the "end product."

SHIELD: A body of material used to prevent or reduce the passage of particles or radiation.

SIEVERT: The SI unit of dose equivalent equal to the absorbed dose in grays multiplied by the quality factor and any other necessary modifying factors.

SOURCE, RADIATION: Materials or devices that make or are capable of making ionizing radiation, including:

a. naturally occuring radioactive materials
b. byproduct materials
c. source materials
d. special nuclear materials
e. fission products
f. materials containing induced or deposited radioactivity
g. radiographic and fluoroscopic equipment
h. particle generators and accelerators
i. electronic equipment that uses klystrons, magnetrons, or other electron tubes that produce x rays.

SOURCE GEOMETRY: The shape, size, and configuration of a radiation source, taken as a whole.

SOURCE MATERIAL: Uranium or thorium or a combination of both, in any physical form, or ores that contain one-twentieth or more by weight of uranium or thorium or any combination. Source material does not include special nuclear material.

SPECIAL NUCLEAR MATERIAL: Plutonium or uranium enriched in isotope 233 or 235, and any other material NRC determines to be special nuclear material. Any material (except source material) artificially enriched by either isotope.

SPECIFIC ACTIVITY: The total activity of a given nuclide per gram of a compound, element, or radioactive nuclide.

SPECIFIC IONIZATION: The number of ion pairs produced per unit path length of ionizing radiation in a medium (e.g., per cm of air or per micron of tissue).

SPECTROMETER (NUCLEAR): A device or instrument, usually electronic, capable of measuring the energy distribution of nuclear radiations.

STABLE ISOTOPE: A nonradioactive isotope of an element.

SURVEY (RADIATION): An evaluation of the radiation hazard associated with the production, use, or existence of radioactive materials or other sources of radiation under specific conditions. The evaluation usually includes:

a. a physical survey of the disposition of materials and equipment
b. measurements or estimates of the levels of radiation involved
c. predictions of hazards resulting from expected or possible changes in materials or equipment.

THIMBLE IONIZATION CHAMBER: A small cylindrical or spherical ionization chamber, usually with walls of organic material.

THRESHOLD DOSE: The minimum absorbed dose that produces a detectable effect.

TISSUE DOSE: The absorbed dose received by tissue in a region of interest, expressed in rads.

TISSUE-EQUIVALENT IONIZATION CHAMBER: An ionization chamber in which the material of the walls, electrodes, and gas are so selected as to produce a response to radiation similar to the response of tissue.

TISSUE-EQUIVALENT MATERIAL: A liquid or solid whose absorbing and scattering properties for a given radiation simulate as closely as possible those of a given biological material, such as fat, bone, or muscle. For muscle or soft tissue, water is usually the best tissue-equivalent material.

TRACK: The visual manifestation of the path of an ionizing particle in a chamber or photographic emulsion.

USE FACTOR: For a mechanical radiation source, the fraction of the workload during which the useful beam is pointed toward the area in question.

USEFUL BEAM: The radiation that passes through the window, aperture, cone, or other collimating device of the housing for a radiation source. Sometimes called "primary beam."

VALENCE: The number representing the combining or displacing power of an atom; the number of electrons lost, gained, or shared by an atom in a compound; the number of hydrogen atoms with which an atom will combine or which it will displace.

VAN DE GRAAFF ACCELERATOR: An electrostatic machine in which electrical charge is carried into the high-voltage terminal by a belt made of an insulating material moving at a high speed. The particles are then accelerated along a discharge path through a vacuum tube by the potential difference between the insulated terminal and the grounded end of the accelerator.

VOLUME, SENSITIVE: The portion of a counter tube or ionization chamber that responds to a specific radiation.

WORKLOAD: A quantity indicating the average weekly output of a mechanical radiation source. For example, for a clinical x-ray apparatus, the workload can be specified in milliampere-minutes per week, at a particular (usually maximum) x-ray tube voltage.

X RAYS: Penetrating electromagnetic radiations whose wavelengths are shorter than those of visible light. They are usually produced by bombarding a metallic target with fast electrons in a high vacuum. In nuclear reactions, it is customary to refer to photons originating in the nucleus as gamma rays, and those originating in the extranuclear part of the atom as x rays. These rays are sometimes called roentgen rays after their discoverer, W. K. Roentgen.

BIBLIOGRAPHY

BIBLIOGRAPHY

ALARA

International Commission on Radiological Protection. 1973. Implications of Commission Recommendations that Doses be Kept as Low as Readily Achievable. ICRP 22, Pergamon Press, New York.

Kathren, R. L., et al. 1980. A Guide to Reducing Radiation Exposures to As Low As Reasonably Achievable (ALARA). DOE/EV/1830-T5, National Technical Information Service, Springfield, Virginia.

Mauro, J. J., and J. Porrovecchio. 1976. "Numerical Criteria for In-Plant 'As Low As Is Reasonably Achievable,'" in Proceedings of the 9th Midyear Topical Symposium of the Health Physics Society on Operational Health Physics. Rocky Mountain Chapter, Health Physics Society, P.O. Box 3229, Boulder, Colorado 80303.

Munson, L. H., and L. A. Freytag. 1976. "An Audible Program of Compliance with ALAP," in Proceedings of the 9th Midyear Topical Symposium of the Health Physics Society on Operational Health Physics. Rocky Mountain Chapter, Health Physics Society, P.O. Box 3229, Boulder, Colorado 80303.

U.S. Nuclear Regulatory Commission. 1975. "Operating Philosophy For Maintaining Occupational Radiation Exposures As Low As Reasonably Achievable." Regulatory Guide 8.1, Washington, D.C.

U.S. Nuclear Regulatory Commission. 1977. "Information Relevant to Ensuring that Occupational Radiation Exposures at Nuclear Power Stations Will be as Low as Reasonably Achievable." Regulatory Guide 8.8, Washington, D.C.

ARMY REGULATIONS

U.S. Department of Defense. Military Standard - Radioactive Materials: Marking and Labeling of Items, Packages, and Shipping Containers for Identification in Use, Storage, and Transportation. MIL-STD-1458, Washington, D.C.

U.S. Department of Defense, Defense Supply Agency. Radioactive Commodities in the DOD Supply Systems. DSAM 4145.8, AR 700-64, Alexandria, Virginia.

U.S. Department of Defense, Defense Supply Agency. Chemical, Biological and Radiological (CBR) Decontamination. TM 3-2300, Washington, D.C.

U.S. Department of the Army, Headquarters. Handling and Disposal of Unwanted Radioactive Material. TM 3-261, Washington, D.C.

U.S. Department of the Army, Headquarters. "Maintenance and Disposition of General Personnel Management and Safety Functional Files." In Army Functional File System. AR 340-11, Washington, D.C.

U.S. Department of the Army, Headquarters. Medical Support - Nuclear/Chemical Accidents and Incidents. AR 40-13, Washington, D.C.

U.S. Department of the Army, Headquarters. Military Traffic Management Regulation. AR 55-355, Washington, D.C.

U.S. Department of the Army, Headquarters. Safety - Accident Reporting and Records. AR 385-40, Washington, D.C.

U.S. Department of the Army, Headquarters. Safety - Ionizing Radiation Protection (Licensing, Control, Transportation, Disposal, and Radiation Safety). AR 385-11, Washington, D.C.

U.S. Department of the Army, Headquarters, Army Materiel Command. Safety - Radiation Protection. DARCOM-R 385-25, Washington, D.C.

U.S. Department of the Army and Defense Lositics Agency. Medical Services - Control and Recording Procedures for Exposure to Ionizing Radiation and Radioactive Materials. AR 40-14, DLAR 1000.28, Washington, D.C.

CONTAMINATION AND DECONTAMINATION

Ayres, J. A. 1970. Decontamination of Nuclear Reactors and Equipment. The Ronald Press Company, New York.

Bost, W. E. 1959. Radioactive Decontamination, A Literature Search. TID-3535, Division of Technical Information Extension, Oak Ridge, Tennessee.

Healy, J. W. 1980. Proposed Policy for Decontamination and Decommissioning. Proceedings of DOE Decommissioning Criteria Workshop in Kansas City, April 15-16, 1980.

International Atomic Energy Agency. 1969. Environmental Contamination by Radioactive Materials. IAEA Publication No. STI/PUB/226, Vienna.

Januska, A. G., W. J. Tyrrell and G. A. Bennett. 1974. Decontamination of Plutonium Contaminated Gloveboxes. ANL-8124, Argonne National Laboratory, Argonne, Illinois.

MacFarlane, C. J., and R. J. Beal. 1962. Special Equipment for Decontamination and Control of Radiation Hazards at Chalk River. ERRISD-1071, Atomic Energy of Canada Limited, Chalk River, Ontario.

Manion, W. J., and Thomas S. LaGuardia. 1980. Decommissioning Handbook. DOE/EV/10128-1, U.S. Department of Energy, Washington, D.C.

Maugham, G. I. 1960. Biological and Radioactive Decontamination of Areas and Equipment. DEG Information Series 147(R), United Kingdom Atomic Energy Association, Risely, Warrington, Lancashire, England.

Philipp, H. 1963. Bibliographic Review of Selected Subjects, Decontamination. German Report AED-C-13-02.

Philipp, H. 1963. Bibliographic Review of Selected Subjects, Decontamination. German Report AED-C-13-03.

Philipp, H. 1966. Bibliographies in Nuclear Science and Technology, Decontamination. German Report AED-C-13-04.

Tompkins, P. C. 1959. "Surface Contamination and Decontamination," in Radiation Hygiene Handbook, 1st ed., H. Blatz, ed. McGraw-Hill, Inc., New York.

DOSE CALCULATIONS

Arras, J. M. 1981. Concentric Cylinder Set Model for Estimating Dose from Gamma-Emitting Cloud. AFRRITN 81-1, Armed Forces Radiobiology Research Institute, Bethesda, Maryland.

Brodsky, A. 1969. "Data and Methods for Estimating Radiation Exposures from Internal and External Radiation Sources," in Handbook of Radioactive Nuclides, Y. Wang, ed. Chemical Rubber Company, Cleveland, Ohio.

Bunch, D. F. 1967. "Dose to Various Body Organs from Inhalation or Ingestion of Soluble Radionuclides," in Handbook of Laboratory Safety, N. V. Steere, ed. Chemical Rubber Company, Cleveland, Ohio.

Cooper, R. E. 1967. RADOS, A Code to Estimate Gamma Dose from a Cloud of Radioactive Gases. DP-1098, Savannah River Laboratory, Aiken, South Carolina.

Cooper, R. E. 1976. "A Computer Code to Provide Emergency Response Capabilities to Estimate Whole Body Gamma Dose," in Savannah River Laboratory Environmental Transport and Effects Research Annual Report, FY-1975. DP-1412, Savannah River Laboratory, Aiken, South Carolina.

Garfield, D. K. 1976. "Estimation of ^{125}Iodine Thyroid Burden from Air Monitoring and Urinalysis Data," in Proceedings of the 9th Midyear Topical Symposium of the Health Physics Society on Operational Health Physics. Rocky Mountain Chapter, Health Physics Society, P.O. Box 3229, Boulder, Colorado 80303.

Healy, J. W., and R. E. Baker. 1968. "Radioactive Clouddose Calculations," in Meteorology and Atomic Energy 1968, D. H. Slade, ed. U.S. Atomic Energy Commission, Washington, D.C.

Houston, J. R., D. L. Strenge and E. C. Watson. 1974. DACRIN: A Computer Program for Calculating Organ Dose from Acute or Chronic Radionuclide Inhalation. BNWL-B-389, Pacific Northwest Laboratory, Richland, Washington.

International Atomic Energy Agency. 1974. Population Dose Evaluation and Standards for Man and His Environent. IAEA Publication No. STI/PUB/375, Vienna.

International Commission on Radiation Units and Measurements. 1964. Physical Aspects of Irradiation. ICRU 106, also published as NBS Handbook 85, U.S. Government Printing Office, Washington, D.C.

International Commission on Radiological Protection. 1960. Report of Committee II on Permissible Dose for Internal Radiation. ICRP 2, Pergamon Press, New York.

International Commission on Radiological Protection. 1968. Report of Committee IV on Evaluation of Radiation Doses to Body Tissues from Internal Contamination Due to Occupational Exposure. ICRP 10, Pergamon Press, New York.

International Commission on Radiological Protection. 1971. Report of Committee V on the Assessment of Internal Contamination Resulting from Recurrent or Prolonged Uptakes. ICRP 10A, Pergamon Press, New York.

International Commission on Radiological Protection. 1972. Alkaline Earth Metabolism in Adult Man. ICRP 20, Pergamon Press, New York.

International Commission on Radiological Protection. 1972. The Metabolism of Compounds of Plutonium and Other Actinides. ICRP 19, Pergamon Press, New York.

International Commission on Radiological Protection. 1978. Limits for Intakes of Radionuclides by Workers. ICRP 30, Part 1 and Supplement to Part 1, Pergamon Press, Oxford.

International Commission on Radiological Protection. 1980. Limits for Intakes of Radionuclides by Workers. ICRP 30, Part 2, Pergamon Press, Oxford.

Jenkins, T. M. 1974. "Accelerator Boundary Doses and Skyshine." Health Phys. 27:251.

Kase, K. R. 1972. Concepts of Radiation Dosimetry. SLAC-153, National Technical Information Service, Springfield, Virginia.

Killough, G. G., and G. R. McKay. 1976. A Methodology for Calculating Radiation Doses from Radioactivity Released to the Environment. ORNL-4992, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

Lahti, G. P., R. S. Hubner and J. C. Golden. 1981. "Assessment of γ-Ray Exposures Due to Finite Plumes." Health Phys. 41:319-340.

Moe, H. J. 1981. "Contact, or Near Contact, External Dose Estimates." Continuing Education Lecture-Health Physics Society Meeting, 1981.

National Council on Radiation Protection and Measurements. 1959. Maximum Permissible Body Burdens and Maximum Permissible Concentration of Radionuclides in Air and in Water for Occupational Exposures. NCRP 22, also published as National Bureau of Standards Handbook 69, Washington, D.C.

Rider, J. L., and S. K. Beal. 1977. Model to Estimate the Local Radiation Doses to Man from the Atmospheric Release of Radionuclides (LWBR Development Program). WAPD-TM-1273, Bettis Atomic Power Laboratory, West Mifflin, Pennsylvania.

Rohrer, R. H. 1970. "Activities of the Medical Internal Radiation Dose Committee (MIRD) of the Society of Nuclear Medicine," in Medical Radionuclides: Radiation Dose and Effects. USAEC Symposium Series No. 20, CONF-691212, National Technical Information Service, Springfield, Virginia.

Scherpelz, R. I., F. J. Borst and G. R. Hoenes. 1980. WRAITH - A Computer Code Which Calculates Internal and External Doses Resulting From Atmospheric Releases of Radioactive Material. PNL-3382, Pacific Northwest Laboratory, Richland, Washington.

Stephens, L. D., R. H. Thomas and S. B. Thomas. 1975. "Population Exposure From High-Energy Accelerators." Health Phys. 29:853.

EMERGENCY PREPAREDNESS

Aldrich, D. C., and J. D. Johnson. 1978. Public Protection Strategies for Potential Nuclear Reactor Accidents: Sheltering Concepts with Existing Public and Private Structures. SAND-77-1725, Massachusetts Institution of Technology, Cambridge, Massachusetts.

American National Standards Institute. 1979. Facilities and Medical Care for On-Site Nuclear Power Plant Radiological Emergencies. ANSI/ANS-3.7.1, American Nuclear Society, LaGrange Park, Illinois.

Benton and Franklin Counties, Washington State. Peace-Time Radiological Training for Firefighting and Paramedic Staff. Benton County, Department of Emergency Services.

Bowers, P. R. 1976. "Emergency Plans - Window Dressing or Workable?" in Proceedings of the 9th Midyear Topical Symposium of the Health Physics Society on Operational Health Physics. Rocky Mountain Chapter, Health Physics Society, P.O. Box 3229, Boulder, Colorado 80303.

Brodsky, A. 1965. "Criteria for Acute Exposure to Mixed Fission-Product Aerosols." Health Phys. 11:1017-1032.

Clarke, R. H. 1973. "The WEERIE Program for Assessing the Radiological Consequences of Airborne Effluents from Nuclear Installations." Health Phys. 25:267.

Dabberdt, W. F., et al. 1982. Workshop on Meteorological Aspects of Emergency Response Plans for Nuclear Power Plants: Proceedings and Recommendations. Final Report, SRI Project 3689, SRI International, Menlo Park, California.

Dickey, B. B. 1976. "The Importance of Coordinated Planning for the Medical Management of Irradiated or Radioactively Contaminated Patients," in Proceedings of the 9th Midyear Topical Symposium of the Health Physics Society on Operational Health Physics. Rocky Mountain Chapter, Health Physics Society, P.O. Box 3229, Boulder, Colorado 80303.

Fitzgerald, J. J. 1970. "Emergency Planning Considerations," page 701 in Applied Radiation Protection and Control, Vol. II. Gordon and Breach Science Publishers, New York.

Gallagher, R. G. 1969. "Emergency Planning and Procedures," in Handbook of Radioactive Nuclides, Y. Wang, ed. Chemical Rubber Company, Cleveland, Ohio.

International Atomic Energy Agency. 1969. Planning for the Handling of Radiation Accidents. IAEA Safety Series No. 32, Vienna.

International Atomic Energy Agency. 1969. Symposium on the Handling of Radiation Accidents. International Atomic Energy Agency, Vienna.

International Atomic Energy Agency. 1974. Evaluation of Radiation Emergencies and Accidents - Selected Criteria and Data. IAEA Technical Report No. 152, Vienna, Austria.

MacDonald, H. F. 1971. "Assessment of Environmental Hazards Following Nuclear Reactor Accidents," in Rapid Methods for Measuring Radioactivity in the Environment. Symposium Proceedings, STI/PUB/289, International Atomic Energy Agency, Vienna, Austria.

McBride, J. A., and R. E. Cunningham. 1972. "The Role of Radiation Monitoring in Emergency Preparedness," in Health Physics Operational Monitoring, Vol. III, C. A. Willis and J. S. Handloser, eds. Gordon and Breach Science Publishers, New York.

Moeller, D. W., and J. M. Selby. 1976. "Planning for Nuclear Emergencies." Nucl. Saf. 17:1.

Montgomery, J. L. 1976. "The Colorado Radiological Response Plan," in Proceedings of the 9th Midyear Topical Symposium of the Health Physics Society on Operational Health Physics. Rocky Mountain Chapter, Health Physics Society, P.O. Box 3229, Boulder, Colorado 80303.

National Council on Radiation Protection and Measurements. 1980. Management of Persons Accidentally Contaminated with Radionuclides. NCRP 65, Washington, D.C.

Pohl, W. K. 1971. "Emergency Organization and Planning." Health Phys. 21:3.

Saenger, E. L. 1963. Medical Aspects of Radiation Accidents: A Handbook for Physicians, Health Physicists, and Industrial Hygienists. U.S. Government Printing Office, Washington, D.C.

Unruh, C. M. 1973. Checklist for Radiological Emergency Preparedness. RS2-158, Pacific Northwest Laboratory, Richland, Washington.

U.S. Department of Energy. 1979. Emergency Handling of Radiation Accident Cases - FIREMEN. DOE/EV-0023, Washington, D.C.

U.S. Department of Energy. 1979. Emergency Handling of Radiation Accident Cases - NURSES. DOE/EV-0018, Washington, D.C.

U.S. Department of Energy. 1979. Emergency Handling of Radiation Accident Cases - PHYSICIANS. DOE/EV-0017, Washington, D.C.

U.S. Department of Energy. 1979. Emergency Handling of Radiation Accident Cases - POLICE. DOE/EV-0021, Washington, D.C.

U.S. Environmental Protection Agency. 1975. Manual of Protective Action Guides and Protective Actions for Nuclear Incidents. EPA-520/1-75-001, revised June 1979 and February 1980, Washington, D.C.

U.S. Nuclear Regulatory Commission (NRC). 1975. Guide and Checklist for Development and Evaluation of State and Local Government Radiological Emergency Response Plans in Support of Fixed Nuclear Facilities. NUREG-71/111, National Technical Information Service, Springfield, Virginia.

U.S. Nuclear Regulatory Commission. 1981. Radiation Protection Plans for Nuclear Power Reactor Licensees. NUREG-0761, U.S. Government Printing Office, Washington, D.C.

U.S. Nuclear Regulatory Commission. 1981. Standard Review Plan for the Review and Evaluation of Emergency Plans for Research and Test Reactors. NUREG-0849, National Technical Information Service, Springfield, Virginia.

Wald, N. 1969. "Radiation Injury and Its Management," in Handbook of Radioactive Nuclides, Y. Wang, ed. Chemical Rubber Company, Cleveland, Ohio.

Washington State Department of Social and Health Services. 1979. Radiation Emergency Handbook, Olympia, Washington.

ENVIRONMENTAL MONITORING

Adams, J. A., W. M. Lowder and T. Gessell, eds. 1972. The Natural Radiation Environment II. CONF-720805, U.S. Energy Research and Development Administration, Washington, D.C.

Cheremisinaff, P. N., and A. C. Morresi. 1977. Environmental Assessment and Impact Statement Handbook. Ann Arbor Science Publishers, Ann Arbor, Michigan.

Corley, J. P., P. H. Denham, D. E. Michels, A. R. Olsen and D. A. Waite. 1976. A Guide for Environmental Radiological Surveillance at ERDA Installations. ERDA-77-24, National Technical Information Service, Springfield, Virginia.

Daly, J. C., and J. J. Gabay. 1971. "Rapid Screening of Environmental Samples Using Gamma Spectrometry." Health Phys. 20:640.

Eisenbud, M. 1973. Environmental Radioactivity. Academic Press, New York.

Hoffman, F. O., C. W. Miller, D. L. Shaeffer and C. T. Garten, Jr. 1977. "Computer Codes for the Assessment of Radionuclides Released to the Environment." Nuclear Safety 18(3):343-354.

Hoy, J. E. 1971. "Environmental Radiation Monitoring with Thermoluminescent Dosimeters." Health Phys. 21:860.

International Atomic Energy Agency. 1966. Environmental Monitoring in Emergency Situations. IAEA Safety Series No. 18, Vienna.

International Atomic Energy Agency. 1966. Manual of Environmental Monitoring in Normal Operations. IAEA Safety Series No. 16, Vienna.

International Atomic Energy Agency. 1975. Objectives and Design of Environmental Monitoring Programmes for Radioactive Contaminations. IAEA Safety Series No. 41, Vienna.

International Atomic Energy Agency. 1978. Monitoring of Airborne and Liquid Radioactive Releases from Nuclear Facilities to the Environment. IAEA Safety Series No. 46, Vienna.

International Atomic Energy Agency. 1978. Principles for Establishing Limits for the Release of Radioactive Releases from Nuclear Facilities to the Environment. IAEA Safety Series No. 45, Vienna.

International Commission on Radiological Protection. 1966. Principles of Environmental Monitoring Related to the Handling of Radioactive Materials. ICRP 7, Pergamon Press, New York.

National Council on Radiation Protection and Measurements. 1976. Environmental Radiation Measurements. NCRP 50, Washington, D.C.

Reinig, W. C., ed. 1970. Environmental Surveillance in the Vicinity of Nuclear Facilities. Charles C. Thomas Publishers, Springfield, Illinois.

U.S. Environmental Protection Agency. 1972. Environmental Radioactivity Surveillance Guide. Report No. ORP/SID 72-2, Washington, D.C.

U.S. Nuclear Regulatory Commission. 1975. "Programs for Monitoring Radioactivity in the Environs of Nuclear Power Plants." Regulatory Guide 4.1, Washington, D.C.

U.S. Nuclear Regulatory Commission. 1975. "Environmental Radiological Monitoring Program." Regulatory Guide 4.8. Washington, D.C.

World Health Organization. 1968. Routine Surveillance for Radionuclides in Air and Water. World Health Organization, Geneva.

FACILITY DESIGN

American Conference of Governmental Industrial Hygienists. 1974. Industrial Ventilation, 13th ed. Committee on Industrial Ventilation, P.O. Box 453, Lansing, Michigan 48902.

American National Standards Institute. 1972. Efficiency Testing of Air Cleaning Systems Containing Devices for Removal of Particles. ANSI N101.1, New York.

Brandt, A. D. 1947. Industrial Health Engineering. John Wiley and Sons, New York.

Burchsted, C. A., A. E. Fuller and J. E. Kahn. 1976. Nuclear Air Cleaning Handbook. ERDA 76-21, U.S. Energy Research and Development Administration, Washington, D.C.

Coco, L. M. 1976. "An Analysis of Facility Design and Operation for Maintaining Personnel Exposures as Low as Reasonably Achievable," in Proceedings of the 9th Midyear Topical Symposium of the Health Physics Society on Operational Health Physics. Rocky Mountain Chapter, Health Physics Society, P.O. Box 3229, Boulder, Colorado 80303.

Hatch, T. 1954. Industrial Dust, 2nd ed. McGraw-Hill, Inc. New York.

Hawkins, M. D. 1959. "Equipment for Handling, Storage, and Transportation of Radioactive Materials," page 17-2 in Radiation Hygiene Handbook, 1st ed., H. Blatz, ed. McGraw-Hill, Inc., New York.

Hemeon, W. C. L. 1954. Plant and Process Ventilation. The Industrial Press, New York.

International Atomic Energy Agency. 1966. Technique for Controlling Air Pollution from the Operation of Nuclear Facilities. IAEA Safety Series No. 17, Vienna.

National Council on Radiation Protection and Measurements. 1977. Radiation Protection Design Guidelines for 0.1-100 MeV Particle Accelerator Facilities. NCRP 51, Washington, D.C.

National Institute for Occupational Safety and Health. 1976. A Guide to Industrial Respiratory Protection. HEW Publication No. 76-189, National Institute for Occupational Safety and Health, Cincinnati, Ohio.

Odor, D. L. 1975. "The Health Physicist and Nuclear Power Plant Design." Health Phys. 29:799.

Patty, F. A., ed. 1958. "Ventilation," in Industrial Hygiene and Toxicology. Interscience Publishers, Inc., New York.

U.S. Nuclear Regulatory Commission. 1975. Design, Testing and Maintenance Criteria for Atmospheric Clean-Up System Air Filtration and Adsorption Units of Light-Water-Cooled Nuclear Power Plants. Regulatory Guide 1.52, Washington, D.C.

Veerling, J. P., Jr., D. G. Oliver, Jr., and E. B. Moore. 1973. "A Storage and Handling Facility for Californium-252 Medical Sources." Health Phys. 25:163.

GENERAL HEALTH PHYSICS PROGRAMS

American National Standards Institute. 1976. Glossary of Terms in Nuclear Science and Technology. ANSI-N1.1, Washington, D.C.

Amtey, S. R., and M. D. Allen. 1976. "Personnel, Space, and Budget Needs of a University Radiation Safety Program," in Proceedings of the 9th Midyear Topical Symposium of the Health Physics Society on Operational Health Physics. Rocky Mountain Chapter, Health Physics Society, P.O. Box 3229, Boulder, Colorado 80303.

Belvin, E. A., and G. F. Stone. 1972. "Management and Health Physics Interaction in a Large Federal Agency," in Health Physics Operational Monitoring, Vol. 3, C. A. Willis and J. S. Handloser, eds. Gordon and Breach Science Publishers, New York.

Boggs, R. F., W. E. Gundaker, W. M. Brobeck and R. G. Altes. 1972. "Health Physics Programs at Low Energy Particle Accelerator Facilities," in Health Physics Operational Monitoring, Vol. 1, C. A. Willis and J. S. Handloser, eds. Gordon and Breach Science Publishers, New York.

Brodsky, A. 1965. "Determining Industrial Hygiene Requirements for Installations Using Radioactive Materials." Amer. Ind. Hyg. Ass. J. 26:294-310.

Ebert, H., H. Eriskat, A. Oudig and G. Uzzan, eds. 1981. Radiation Protection Optimization, Present Experience and Methods. Pergamon Press, New York.

Evans, E. A., M. Muranotsu, et al. 1977. Radiotracer Techniques and Applications. Marcel Dekker, Inc., New York.

Fitzgerald, J. J. 1969. Applied Radiation Protection and Control. Gordon and Breach Science Publishers, New York.

Ice, R. D. 1971. "Establishment of a University Radiation Safety Office." Health Phys. 20:444.

International Atomic Energy Agency. 1965. Provision of Radiological Protection Services. IAEA Safety Series No. 13, Vienna.

International Atomic Energy Agency. 1973. Radiation Protection Procedures. IAEA Safety Series No. 38, Vienna.

International Atomic Energy Agency. 1973. Safe Handling of Radionuclides. IAEA Safety Series No. 1, Vienna.

International Commission on Radiological Protection. 1971. Data for Protection Against Ionizing Radiation from External Sources. ICRP 21, supplement to ICRP 15. Pergamon Press, New York.

Lindell, B. 1971. "Professional Responsibilities of the Health Physicist in Relation to the Medical Profession." Health Phys. 20:475.

McConnon, D. 1972. "A Health Physics Program for a Plutonium Fuel Fabrication Facility," in Health Physics Operational Monitoring, Vol. 1, C. A. Willis and J. S. Handloser, eds. Gordon and Breach Science Publishers, New York.

Miller, K. L. 1976. "A Radiation Safety Program for a New Medical Center," in Proceedings of the 9th Midyear Topical Symposium of the Health Physics Society on Operational Health Physics. Rocky Mountain Chapter, Health Physics Society, P.O. Box 3229, Boulder, Colorado 80303.

National Council on Radiation Protection and Measurements. 1971. Basic Radiation Protection Criteria. NCRP 39, NCRP Publications, Washington, D.C.

National Council on Radiation Protection and Measurements. 1971. Protection Against Neutron Radiation. NCRP 38. Washington, D.C.

National Council on Radiation Protection and Measurements. 1978. Operational Radiation Safety Program. NCRP 59, Washington, D.C.

Norwood, W. D. 1975. Health Protection of Radiation Workers. Charles C. Thomas, Publisher, Springfield, Illinois.

Patterson, H. W., and R. H. Thomas. 1973. Accelerator Health Physics. Academic Press, New York.

Remley, M. E. 1972. "Management Requirements on Health Physics," in Health Physics Operational Monitoring, Vol. 3, C. A. Willis and J. S. Handloser, eds. Gordon and Breach Science Publishers, New York.

Tolen, J. H. 1972. "Organization of a Small-Scale Radiation Safey Program," in Health Physics Operational Monitoring, Vol. 1, C. A. Willis, and J. S. Handloser, eds. Gordon and Breach Science Publishers, New York.

Unruh, C. M. 1972. Problems for Radiation Protection Specialists for Safety Evaluations. BNWL-SA-4459, Pacific Northwest Laboratory, Richland, Washington.

U.S. Code of Federal Regulations. 1982. Title 10, Parts 0 to 199. U.S. Government Printing Office, Washington, D.C.

U.S. Code of Federal Regulations. 1982. Title 10, Part 20, "Standards for Protection Against Radiation." Washington, D.C.

U.S. Code of Federal Regulations. 1982. Title 10, Part 30, "Rules of General Applicability to Licensing of Byproduct Material." Washington, D.C.

GENERAL TEXTBOOKS

Adams, A. S., and W. M. Lowder, eds. 1964. The Natural Radiation Environment. University of Chicago Press, Chicago, Illinois.

Arena, V. 1971. Ionizing Radiation and Life. The C. B. Mosby Company, St. Louis, Missouri.

Attix, F. H., E. Tochilin and W. C. Roesch, eds. 1968. Fundamentals. Vol. 1 of Radiation Dosimetry, 2nd ed. Academic Press, New York.

Attix, F. H., E. Tochilin and W. C. Roesch, eds. 1968. Instrumentation, Vol. 2 of Radiation Dosimetry, 2nd ed. Academic Press, New York.

Blatz, H. 1964. Introduction to Radiological Health. McGraw-Hill, New York.

Bushong, S. C. 1975. Radiological Science for Technologists. The C.V. Mosby Company, St. Louis, Missouri.

Cember, H. 1969. Introduction to Health Physics. International Series of Monographs in Nuclear Energy, Volume 105. Pergamon Press, New York.

Eisenbud, M. 1973. Environmental Radioactivity, 2nd ed. Academic Press, New York.

Fitzgerald, J. J. 1969. Applied Radiation Protection and Control, Vol. I and II. Gordon and Breach Science Publishers, New York.

Fitzgerald, J. J., G. L. Brownell and F. J. Mahoney. 1967. Mathematical Theory of Radiation Dosimetry. Gordon and Breach Science Publishers, New York.

Gloyna, E. F., and J. O. Ledbetter. 1969. Principles of Radiological Health. Marcel Dekker, Inc., New York.

Hall, E. J. 1978. Radiobiology for the Radiologist, 2nd ed. Harper and Row, Hagerstown, Maryland.

Hendee, W. R. 1970. Medical Radiation Physics. Year Book Medical Publishers, Inc., Chicago, Illinois.

Henry, H. F. 1969. Fundamentals of Radiation Protection. Wiley-Interscience, New York.

Hine, J. G., and G. L. Brownell, eds. 1956. Radiation Dosimetry. Academic Press, New York.

Johns, H. E., and J. R. Cunningham. 1974. The Physics of Radiology, 3rd ed. Charles C. Thomas, Springfield, Illinois.

Knoll, G. F. 1979. Radiation Detection and Measurement. John Wiley and Sons, New York.

Lapp, R. E., and H. L. Andrews. 1972. Nuclear Radiation Physics, 4th ed. Prentice-Hall, Englewood Cliffs, New Jersey.

Lederer, C.M., and V. S. Shirley, eds. 1978. Table of Isotopes. John Wiley and Sons, New York.

Moe, J. J., S. R. Lusuk, M. C. Shumacher and H. M. Hunt. 1972. Radiation Safety Technician Training Course. ANL-7291, Rev. 1, Argonne National Laboratory, Argonne, Illinois.

Morgan, K. Z., and J. E. Turner, eds. 1967. Principles of Radiation Protection. John Wiley and Sons, New York.

Norwood, W. D. 1975. Health Protection of Radiation Workers. Charles C. Thomas, Springfield, Illinois.

Radiological Health Handbook. 1970. U.S. Department of Health, Education and Welfare, Bureau of Radiological Health. U.S. Government Printing Office, Washington, D.C.

Shapiro, J. 1972. Radiation Protection. Harvard University Press, Cambridge, Massachusetts.

Shapiro, J. 1981. Radiation Protection, A Guide for Scientists and Physicians. Harvard University Press, Cambridge, Massachusetts.

INSTRUMENTATION AND DOSIMETRY

American National Standards Institute. 1972. Criteria for Film Badge Performance. ANSI N13.7, New York.

American National Standards Institute. 1975. Performance, Testing, and Procedural Specifications for Thermoluminescent Dosimetry: Environmental Applications. ANSI N5.45, New York.

American National Standards Institute. 1978. Radiation Protection Instrumentation Test and Calibration. ANSI N323, The Institute of Electrical and Electronics Engineers, Inc., New York.

Apt, K. E., and K. J. Schiager. 1975. "A Passive Environmental Neutron Dosimeter." Health Phys. 28:474.

Becker, K. 1972. "The Future of Personnel Dosimetry." Health Phys. 23:729.

Boyns, P. K. 1976. The Aerial Radiological Measuring System (ARMS)--Systems, Procedures and Sensitivity. EGG 1183-1691, EG&G Energy Measurements Group, Las Vegas, Nevada.

Brackenbush, L. W., et al. 1980. Personnel Neutron Dosimetry at Department of Energy Facilities. PNL-3213, Pacific Northwest Laboratory, Richland, Washington.

Brodsky, A. 1969. "Personnel Dosimetry," in Handbook of Radioactive Nuclides, Y. Wang, ed. Chemical Rubber Company, Cleveland, Ohio.

Budnitz, R. J. 1974. "Radon-222 and Its Daughters - A Review of Instrumentation for Occupational and Environmental Monitoring." Health Phys. 26:145.

Budnitz, R. J. 1974. "Tritium Instrumentation for Environmental and Occupational Monitoring-A Review." Health Phys. 26:165.

Dudley, R. A. 1966. "Dosimetry with Photographic Emulsions," in Radiation Dosimetry, Vol. IId, F. H. Attix, W. C. Roesch, and E. E. Tochilin, eds. Academic Press, New York.

Eichholz, G. G., and J. W. Poston. 1979. Principles of Nuclear Radiation Detection. Ann Arbor Science Publishers, Inc., Ann Arbor, Michigan.

European Nuclear Energy Agency. 1963. Proceedings of a Symposium on Personnel Dosimetry Techniques for External Radiation. McGraw-Hill, Inc., New York.

Fitzgerald, J. J. 1969. "Instruments for Radiation Detection and Measurement," in Applied Radiation Protection and Control, Vol. I. Gordon and Breach Science Publishers, New York.

Gessell, T. F., G. de P. Burke and K. Becker. 1976. "An International Intercomparison of Environmental Dosimeters." Health Phys. 30:125.

Gibbs, W. D., and C. C. Lushbaugh. 1969. "Whole-Body Counter System," in Handbook of Radioactive Nuclides, Y. Wang, ed. Chemical Rubber Company, Cleveland, Ohio.

Hankins, D. E. 1973. "Progress in Personnel Neutron Dosimetry," in Proceedings of the 3rd International Congress of the International Radiation Protection Association." CONF-730907, National Technical Information Service, Springfield, Virginia.

Hart, J. C. 1972. "Legal and Administrative Aspects of Personnel Dosimetry." Health Phys. 23:343.

Howard, L. E. Jr., J. H. Spickard and M. Wilhelmsen. 1971. "A Human Radioactivity Counter and Medical Van." Health Phys. 21:417.

Hoy, J. E. 1972. "An Albedo-Type Personnel Neutron Dosimeter." Health Phys. 24:385.

International Atomic Energy Agency. 1970. Nuclear Accident Dosimetry Systems. IAEA Publication No. STI/PUB/241, Vienna.

International Atomic Energy Agency (IAEA). 1970. Personnel Dosimetry Systems for External Radiation Exposures. IAEA Publication No. 109, Vienna.

Joanes, A. P. 1975. "A Personal Contamination Monitor Using Large Area Geiger Counters." Health Phys. 28:521.

Kocher, L. F., L. L. Nichols, G. W. R. Endres, D. V. Shipler and A. J. Haverfield. 1973. "The Hanford Thermoluminescent Multipurpose Dosimeter." Health Phys. 25:567.

Knoll, G. F. 1979. Radiation Detection and Measurements. John Wiley and Sons, New York.

National Council on Radiation Protection and Measurements. 1978. Instrumentation and Monitoring Methods for Radiation Protection. NCRP 57. Washington, D.C.

Oshino, M. 1973. "Response of NTA Personnel Neutron Monitoring Film Worn on Human Phantom." Health Phys. 24:71.

Price, W. J. 1964. Nuclear Radiation Detection. 2nd ed. McGraw-Hill Inc., New York.

Rich, B. L., and B. G. Samardzich. 1973. "Organizational Aspects of a Personnel Dosimetry Service." Health Phys. 24:95.

Sommers, J. F. 1975. "Sensitivity of the G-M and Ion Chamber Beta-Gamma Survey Instruments." Health Phys. 28, 755.

Technical Education Research Center-S.W. 1980. Course II, Radiation Detection and Measurements. Waco, Texas.

U.S. Department of Defense, Dense Civil Preparedness Agency. 1977. Handbook for Aerial Radiological Monitors. CPG-2-6.2.3, Washington, D.C.

U.S. Nuclear Regulatory Commission (NRC). 1973. "Film Badge Performance Criteria." Regulatory Guide 8.3, Washington, D.C.

QUALITY ASSURANCE/QUALITY CONTROL

American National Standards Institute. 1973. Quality Assurance Terms and Definitions. ANSI N45.2.10, New York.

American National Standards Institute. 1974. Quality Assurance Requirements for the Design of Nuclear Power Plants. ANSI N45.2.11, New York.

American National Standards Institute. 1976. Administration Controls and Quality for Operational Place of Nuclear Power Plants. ANSI N18.7, New York.

American National Standards Institute and American Society of Mechanical Engineers. 1979. Quality Assurance Program Requirements for Nuclear Facilities. ANSI/ASME N45.2, New York.

International Atomic Energy Agency. 1979. Quality Assurance in the Procurement of Items and Services for Nuclear Power Plants. IAEA Safety Series No. 50-SG-QA3, Vienna.

International Atomic Energy Agency. 1979. Quality Assurance Records Systems. IAEA Safety Series No. 50-SG-QA2, Vienna.

U.S. Atomic Energy Commission, Division of Reactor Research and Development. 1973. Quality Assurance Program Requirements. RDT Standard F2-2, Washington, D.C.

U.S. Code of Federal Regulations. 1982. Title 10, Part 50, Appendix B, "Quality Assurance Criteria for Nuclear Power Plants and Fuel Reprocessing Plants." Washington, D.C.

U.S. Nuclear Regulatory Commission. 1974. "Quality Assurance Terms and Definitions." Regulatory Guide 1.74, Washington, D.C.

U.S. Nuclear Regulatory Commission. 1975. "Quality Assurance Requirements for the Design of Nuclear Power Plants." Regulatory Guide 1.64, Rev. 1, Washington, D.C.

U.S. Nuclear Regulatory Commission. 1979. "Quality Assurance for Radiological Monitoring Programs (Normal Operations) - Effluent Streams and the Environment." Regulatory Guide 4.15, Rev. 1, Washington, D.C.

U.S. Nuclear Regulatory Commission. 1979. "Quality Assurance Program Requirements (Operation)." Proposed Revision 3 of Regulatory Guide 1.33, Washington, D.C.

U.S. Nuclear Regulatory Commission. 1980. "Auditing of Quality Assurance Programs for Nuclear Power Plants." Regulatory Guide 1.144, Rev. 1, Washington, D.C.

RADIATION-GENERATING DEVICES

American National Standards Institute. 1967. Immediate Evacuation Signal for Use in Industrial Installations Where Radiation Exposure May Occur. ANSI N2.3, New York.

American National Standards Institute. 1975. General Safety Standard for Installations Using Non-Medical X-Ray and Sealed Gamma-Ray Sources, Energies up to 10 MeV. ANSI N543, also published as National Bureau of Standards Handbook No. 114, Washington D.C.

American National Standards Institute. 1976. Radiological Safety Standards for the Design of Radiographic and Fluoroscopic Industrial X-Ray Equipment. ANSI N537, New York.

American National Standards Institute. 1977. Radiation Safety for X-Ray Diffraction and Fluorescence Analysis Equipment. ANSI No. N43.2, also published as National Bureau of Standards Handbook No. 111, Washington, D.C.

Devanney, J. A., and C. J. Daniels. 1976. "Radiation Leakage from Electron Microscopes." Health Phys. 20:231.

International Atomic Energy Commission. 1979. Radiological Safety Aspects of the Operation of Electron Linear Accelerators. Vienna.

National Council on Radiation Protection and Measurements. 1968. Medical X-Ray and Gamma-Ray Protection for Energies up to 10 MeV - Equipment Design and Use. NCRP 33, Washington D.C.

RADIOCHEMICAL ANALYSIS

International Atomic Energy Agency. 1966. Quick Methods for Radiochemical Analysis. IAEA Technical Report No. 95, Vienna.

RECORDS

American National Standards Institute. 1972. Practice for Occupational Radiation Exposure Records Systems. ANSI N13.6-1966, Rev. 1972, New York.

Boiter, H. P. 1976. "Radiation Exposure Records Management," in Proceedings of the 9th Midyear Topical Symposium of the Health Physics Society on Operational Health Physics. Rocky Mountain Chapter, Health Physics Society, P.O. Box 3229, Boulder, Colorado 80303.

Eastman Kodak Company. 1978. Storage and Preservation of Microfilms. Kodak Pamphlet D-31, Rochester, New York.

Matterazzi, A. R. 1978. Archival Stability of Microfilm - A Technical Review. Technical Report No. 18, U.S. Government Printing Office, Washington, D.C.

U.S. Department of Energy. 1980. "Micrographics Management." DOE Order 1300.1, Washington, D.C.

SHIELDING

Blizard, E. P., and L. S. Abbot, eds. 1962. Reactor Handbook, 2nd ed., Vol. III, Part B, "Shielding." Interscience Publishers, Inc., New York.

Borak, T. B. 1975. "A Simple Approach to Calculating Gamma-Ray Skyshine for Reduced Shielding Applications." Health Phys. 29:423.

Bozyap, O., and L. R. Day. 1975. "Attenuation of 15 MeV Neutrons in Shields of Concrete and Paraffin Wax." Health Phys. 28:101.

Burson, Z. G., and A. E. Profio. 1975. Structure Shielding from Cloud and Fallout Gamma Ray Sources for Assessing the Consequences of Reactor Accidents. EGG-1183-1670, EG&G, Inc., Las Vegas, Nevada.

DeAlmeida, C. E., S. A. Rosanky, J. R. Marbach and P. R. Almond. 1975. "Transmission in Concrete and Scatter Angular Distribution of 25-MV X Rays from a Betatron and a Linear Accelerator." Health Phys. 28:771.

Jaeger, R. G., ed. 1968. Engineering Compendium on Radiation Shielding - Fundamentals and Methods, Vol. I. Springer-Verlag, New York.

Jaeger, R. G., ed. 1970. Engineering Compendium on Radiation Shielding - Shielding Design and Engineering, Vol. III. Springer-Verlag, New York.

Jaeger, R. G., ed. 1975. Engineering Compendium on Radiation Shielding - Shielding Materials, Vol. II. Springer-Verlag, New York.

Jones, T. D., and F. F. Haywood. 1975. "Transmission of Photons Through Common Shielding Media." Health Phys. 28:630.

National Council on Radiation Protection and Measurements. 1976. Structural Shielding Design and Evaluation for Medical Use of X-Rays and Gamma Rays of Energies Up to 10 MeV. NCRP 49, Washington, D.C.

Patterson, H. W., and R. H. Thomas. 1973. "Accelerator Shielding," in Accelerator Health Physics. Academic Press, New York.

Price, B. T., C. C. Horton and K. T. Spinney. 1957. Radiation Shielding. Pergamon Press, New York.

Rockwell, T., III, ed. 1956. Reactor Shielding Design Manual, 1st ed. Van Nostrand, Princeton.

Schaeffer, N. M., ed. 1973. Reactor Shielding for Nuclear Engineering, TID-25951, National Technical Information Service, Springfield, Virginia.

Trout, E. D., J. P. Kelley and G. L. Herbert. 1975. "X-Ray Attenuation in Steel-50 to 300 kVp." Health Phys. 29:163.

U.S. Nuclear Regulatory Commission. 1975. "Concrete Radiation Shields for Nuclear Power Plants." Regulatory Guide 1.69. Washington, D.C.

STATISTICS

Ostle, B., and R. W. Mensing. 1975. Statistics in Research, 3rd. ed. Iowa State University Press, Iowa.

Speer, D. R., and D. A. Waite. 1976. "Statistical Distributions as Applied to Environmental Surveillance Data," in Proceedings of the 9th Midyear Topical Symposium of the Health Physics Society on Operational Health Physics. Rocky Mountain Chapter, Health Physics Society, P.O. Box 3229, Boulder, Colorado 80303.

SURVEYING AND MONITORING

American National Standards Institute. 1969. Administrative Practices in Radiation Monitoring (A Guide for Management). ANSI N13.2, New York.

American National Standards Institute. 1969. Guide for Administrative Practices in Radiation Monitoring. ANSI N13.2, New York.

Diegl, H. 1972. "Guidelines for Determining Frequency of Wipe Surveys," in Health Physics Operational Monitoring, Vol. I, C. A. Willis and J. S. Handloser, eds. Gordon and Breach Science Publishers, New York.

Feldman, A. 1976. "Factors and Strategies in Selection of Instruments for Radiation Surveys Around Medical X-Ray Installations," in Proceedings of the 9th Midyear Topical Symposium of the Health Physics Society on Operational Health Physics. Rocky Mountain Chapter, Health Physics Society, P.O. Box 3229. Boulder, Colorado 80303.

Fish, B. R., ed. 1964. Surface Contamination. Pergamon Press, New York.

Gaeta, N. A., and M. H. Repacholi. 1975. "A Standard Survey Procedure for Photofluorographic X-Ray Machines." Health Phys. 28:763.

Iacovino, J. M. 1976. "Radiation Monitoring Systems in a Changing Regulatory Environment," in Proceedings of the 9th Midyear Topical Symposium of the Health Physics Society on Operational Health Physics. Rocky Mountain Chapter, Health Physics Society, P.O. Box 3229. Boulder, Colorado 80303.

International Atomic Energy Agency. 1979. Radiological Surveillance of Airborne Contaminants in the Working Environment. IAEA Safety Series No. 49, Vienna.

International Atomic Energy Association. Particle Size Analysis in Estimating the Significance of Airborne Contamination. IAEA Safety Series No. 179, Vienna.

Katz, M. 1977. Methods of Air Sampling and Analysis. R. R. Donnelley and Sons Company, Crawfordsville, Indiana.

McClelland, T. W., and E. D. McFall. 1976. "Radiation Monitoring Considerations for Radiobiology Facilities," in Proceedings of the 9th Midyear Topical Symposium of the Health Physics Society on Operational Health Physics. Rocky Mountain Chapter, Health Physics Society, P.O. Box 3229, Boulder, Colorado 80303.

Moghissi, A. A., and M. W. Carter. 1973. Tritium. Messenger Graphics, Phoenix, Arizona.

National Council on Radiation Protection and Measurements. 1976. Tritium Measurement Techniques. NCRP 47, Washington, D.C.

National Council on Radiation Protection and Measurements. 1978. A Handbook of Radioactivity Measurement Procedures. NCRP 58, Washington, D.C.

National Council on Radiation Protection and Measurements. 1978. Instrumentation and Monitoring Methods for Radiation Protection. NCRP 57, Washington, D.C.

Noll, K. E., and T. L. Miller. 1977. Air Monitoring Survey Design. Ann Arbor Science Publishers Inc., Ann Arbor, Michigan.

Olson, O. L. 1976. "A Determination of Criteria for a Bioassay Program," in Proceedings of the 9th Midyear Topical Symposium of the Health Physics Society on Operational Health Physics. Rocky Mountain Chapter, Health Physics Society, P.O. Box 3229, Boulder, Colorado 80303.

Orvis, A. L. 1970. "Whole Body Counting," in Medical Radionuclides: Radiation Dose and Effects. USAEC Symposium Series No. 20, CONF-691212, National Technical Information Service, Springfield, Virginia.

Unruh, C. M. 1970. "Radiation Protection Practices for Tritium - A Manual of Good Practice." BNWL-SA-3390, Pacific Northwest Laboratory, Richland, Washington.

U.S. Nuclear Regulatory Commission. 1975. "Acceptable Concepts, Models, Equations and Assumptions for a Bioassay Program.J Regulatory Guide 8.9. Washington, D.C.

Wade, J. E., and G. E. Cunningham. 1967. Radiation Monitoring: A Programmed Instruction Book. Division of Technical Information, Oak Ridge National Laboratory, Oak Ridge, Tennessee.

TRAINING

American National Standards Institute (ANSI). American National Standard for Selection and Training of Nuclear Power Plant Personnel. ANSI/ANS-3.1, American Nuclear Society, LaGrange Park, Illinois.

American National Standards Institute. 1979. Proposed American National Standard for the Qualification and Training of Personnel For Nuclear Power Plants. ANSI/ANS-3.1, American Nuclear Society, LaGrange Park, Illinois.

American National Standards Institute. 1980. Practices for Respiratory Protection. ANSI-Z88.2, New York.

American Nuclear Society (ANS). 1976. Proposed American National Standard for the Qualification and Training of Personnel for Research Reactors. ANS-15.4, LaGrange Park, Illinois.

Brown, B. 1971. "A New 2-Year Part-Time Modular MSc Source in Health Physics at the University of Salford." Health Phys. 20:663.

International Atomic Energy Agency. 1964. Training in Radiological Protection: Curricula and Programming. IAEA Technical Report No. 31, Vienna.

Medina, L. C., W. D. Kittinger and R. M. Vogel. 1976. "Radiation Monitor Training Program at Rocky Flats," in Proceedings of the 9th Midyear Topical Symposium of the Health Physics Society on Operational Health Physics. Rocky Mountain Chapter, Health Physics Society, P.O. Box 3229, Boulder, Colorado 80303.

National Council on Radiation Protection and Measurements (NCRP). 1978. Radiation Safety Training Criteria for Industrial Radiography. NCRP 61, Washington, D.C.

U.S. Code of Federal Regulations. 1982. Title 10, Part 19, "Notices, Instructions and Reports to Workers; Inspections." Washington, D.C.

U.S. Nuclear Regulatory Commission. 1979. Information Relevant to Ensuring That Occupational Radiation Exposures at Nuclear Power Stations Will Be As Low As Reasonably Achievable (ALARA)." Regulatory Guide 8.8, Washington, D.C.

U.S. Nuclear Regulatory Commission. 1980. "Personnel Qualification and Training." Second Proposed Revision to Regulatory Guide 1.8, Washington, D.C.

U.S. Nuclear Regulatory Commission. 1981. "Instruction Concerning Risks From Occupational Radiation Exposure." Regulatory Guide 8.29, Washington, D.C.

U.S. Nuclear Regulatory Commission. 1981. "Radiation Protection Training for Personnel At Light-Water-Cooled Nuclear Power Plants." Regulatory Guide 8.27, Washington, D.C.

Vetter, R. J., and P. L. Ziemer. 1976. "Operational Training in the Health Physics Curriculum," in Proceedings of the 9th Midyear Topical Symposium of the Health Physics Society on Operational Health Physics. Rocky Mountain Chapter, Health Physics Society, P.O. Box 3229, Boulder, Colorado 80303.

Williams, S. L. 1976. "Meaningful Radiation Worker Training for Temporary Craftsmen," in Proceedings of the 9th Midyear Topical Symposium of the Health Physics Society on Operational Health Physics. Rocky Mountain Chapter, Health Physics Society, P.O. Box 3229, Boulder, Colorado 80303.

TRANSPORTATION

American National Standards Institute. 1973. Administrative Guide for Packaging and Transporting Radioactive Materials. ANSI N14.10.1, New York.

American National Standards Institute. 1975. Administrative Guide for Verifying Compliance with Packaging Requirements for Shipments of Radioactive Materials. ANSI N14.10.3, New York.

International Atomic Energy Agency. 1979. Regulations for the Safe Transport of Radioactive Materials. IAEA Safety Series No. 6, Vienna.

U.S. Code of Federal Regulations. 1982. Title 49, Parts 100-199. Washington, D.C.

U.S. Code of Federal Regulations. 1982. Title 10, Part 71, "Packaging of Radioactive Materials for Transport and Transportation of Radioactive Materials Under Certain Conditions." Washington, D.C.

U.S. Department of Transportation. 1977. A Review of the Department of Transportation (DOT) Regulations for Transportation of Radioactive Materials. Washington, D.C.

U.S. Nuclear Regulatory Commission. 1975. "Administrative Guide for Obtaining Exemptions from Certain NRC Requirements over Radioactive Materials Shipment." Regulatory Guide 7.5, Washington, D.C.

U.S. Nuclear Regulatory Commission. 1975. "Leakage Tests on Packages for Shipment of Radioactive Materials." Regulatory Guide 7.4, Washington, D.C.

U.S. Nuclear Regulatory Commission. 1975. "Procedures for Picking Up and Receiving Packages of Radioactive Material." Regulatory Guide 7.3, Washington, D.C.

U.S. Nuclear Regulatory Commission. 1977. "Administrative Guide for Verifying Compliance with Packaging Requirements for Shipment of Radioactive Materials." Regulatory Guide 7.7, Washington, D.C.

WASTE MANAGEMENT

Bradley, F. J. 1969. "Radioactive Waste Disposal," in Handbook of Radioactive Nuclides, Y. Wang, ed. Chemical Rubber Company, Cleveland, Ohio.

Cardozo, R. L. 1973. "The Dispersal of Radioactive Matter by Evaporation." Health Phys. 25:593.

Gallagher, F. E., III. 1976. "A New Facility for Processing and Storage of Radioactive and Toxic Chemical Waste," in Proceedings of the 9th Midyear Topical Symposium of the Health Physics Society on Operational Health Physics. Rocky Mountain Chapter, Health Physics Society, P.O. Box 3229, Boulder, Colorado 80303.

Gera, F. 1974. "The Classification of Radioactive Waste." Health Phys. 27:113.

Gregory, W. D., and H. D. Maillie. 1975. "Incinceration of Animal Radioactive Waste: A Comparative Cost Analysis." Health Phys. 29:389.

International Atomic Energy Agency. 1965. Management of Radioactive Wastes Produced by Radioisotope Users. IAEA Safety Series No. 12, Vienna.

International Atomic Energy Agency. 1966. Management of Radioactive Wastes Produced by Radioactive Users-Technical Addendum. IAEA Safety Series No. 19, Vienna.

Mawson, C. A. 1965. Management of Radioactive Waste. Van Nostrand, New York.

Murphy, P. H., and N. S. Anderson. 1975. "Inexpensive, Convenient Xenon Disposal." Health Phys. 29:779.

Port, E. A. 1975. "An Improved Receptacle for Radioactive Waste." Health Phys. 29:801.

U.S. Nuclear Regulatory Commission. 1975. "Evaluating and Reporting Radioactivity in Solid Wastes and Releases of Radioactive Materials in Liquid and Gaseous Effluents from Light-Water-Cooled Nuclear Power Plants." Regulatory Guide 1.21, Washington, D.C.

DISTRIBUTION

No. of
Copies

OFFSITE

2 DOE Technical Information Center

ONSITE

32 Pacific Northwest Laboratory

J. L. Baer
L. W. Brackenbush
E. H. Carbaugh
M. Cross
T. H. Essig
L. G. Faust (3)
R. L. Gilchrist (6)
W. A. Glass
D. E. Hadlock
W. N. Herrington
J. C. Mann
L. H. Munson
B. L. Murphy (3)
D. W. Murphy
J. M. Selby
S. L. Stein
J. E. Tanner
J. M. Taylor
R. J. Traub
C. M. Unruh
Radiological Sciences Library
Technical Information (2)