chapter

23

Radiation Safety and Health Physics

Chapter 22 dealt with the radiation dose received by patients undergoing nuclear medicine procedures. This chapter deals primarily with the exposure of personnel who work in nuclear medicine clinics and research laboratories and who are exposed to radiation in their normal working environment. Stored radioactive materials, handling of calibration sources, preparation of radioactive materials for patients and phantoms, and proximity to patients or phantoms to whom these preparations have been administered all are potential sources of radiation exposure. An additional problem is the potential for radiation exposure to nonlaboratory personnel, such as patient relatives, attending nursing staff, and even passers-by in the hallways adjacent to the laboratory.

The quantities of radioactive material used and radiation levels encountered in a nuclear medicine laboratory generally are well below what is necessary to cause any type of "radiation sickness." Of more concern are the long-term effects that may possibly result from chronic exposures to even low levels of radiation. The most important of these effects are genetic damage to cells (mutagenesis), damage to chromosomes (clastogenesis), and carcinogenesis.

Presently, our understanding of the effects of chronic exposure to low levels of radiation is far from complete. Radiation protection regulations and guidelines currently are based on a *linear nonthreshold* (*LNT*) *model,* which assumes that there is no "threshold dose" for these long-term effects and that the risk increases linearly with radiation dose.[1] There also are experiments, data, and proposed radiation injury models that are inconsistent with the LNT model.[2,3] Some scientists argue that studies involving low levels of radiation suggest that low doses actually have a beneficial effect on health, resulting from stimulation of the immune system.[4] This effect is known as *radiation hormesis.* A vigorous debate about the biologic consequences of low levels of ionizing radiation, the relevance of absorbed dose estimates in assessing health risks, and the effect of these findings on regulations pertaining to radiation exposure is likely to continue for some years to come. Whatever the outcome of this debate, and even though the risks to personnel occupationally exposed to ionizing radiation in the nuclear medicine environment clearly are small (based on decades of historic data), common sense dictates that radiation exposures in and around the nuclear medicine laboratory be kept as low as is reasonably achievable.

When considering possible health effects to nuclear medicine patients or occupationally exposed personnel, it also is important to place the dose received in perspective by considering the radiation dose received by all of us from natural background sources. These sources include naturally occurring radionuclides in the body (e.g., ^{40}K), cosmic radiation, and radionuclides that occur naturally in the environment. Effective doses to individuals per year from these natural sources average approximately 2.4 mSv (typical range 1-13 mSv).[5] As shown in Example 22-10, a 250-MBq injection of ^{18}F-fluorodeoxyglucose leads to an effective dose of roughly 5.8 mSv (equal to the dose that would be received in approximately 1.7 years from nature). The average effective dose to the extremities of nuclear medicine technical personnel is on the order of 4 mSv per year.[6]

The analysis of problems in the handling of radiation sources and the development of safe handling practices are the general concerns of the broad field of *health physics.* The

practices that are prescribed by this analysis are sometimes expressed formally as regulations and sometimes as "common sense" recommendations. In this chapter we primarily discuss aspects of health physics and radiation safety practices as they apply to the nuclear medicine laboratory. However, a further responsibility arises because nuclear medicine scientists and practitioners often are among the first people contacted (e.g., by the media) for information on public-health radiation issues. Therefore it is wise to know where reliable sources of information can be found. A number of international organizations such as the International Commission on Radiological Protection (ICRP), the United Nations Scientific Committee on the Effects of Atomic Radiation and the International Atomic Energy Agency provide useful reports and literature. Selected references and websites are provided at the end of this chapter.

A. QUANTITIES AND UNITS

1. Dose-Modifying Factors

For health physics purposes, specification of the radiation absorbed dose in grays (see Chapter 22, Section A) is inadequate for a complete and accurate assessment of potential radiation hazards. Although the relative risk of potential injury increases with increasing absorbed dose values, several other *dose-modifying factors* also must be taken into account.

1. *The part of the body exposed.* Total-body exposure carries a greater risk than partial-body exposure. Exposure of major organs in the trunk of the body is more serious than exposure to the extremities. The active blood-forming organs, the gonads, and the lens of the eye are especially sensitive to radiation damage. A superficial dose to the skin (e.g., from an external source of β particles) is less hazardous than the same dose delivered to greater depths (e.g., from an external source of γ rays or from internally deposited radioactivity).
2. *The time span over which the radiation dose is delivered.* A given number of grays delivered over a short period (e.g., minutes or hours) has a greater potential for damage than the same dose delivered over a long period (e.g., months or years).
3. *The age of the exposed individual.* Children are more susceptible to injurious radiation effects than are adults. The developing embryo and fetus are especially sensitive.
4. *The type of radiation involved.* In general, densely ionizing radiation [i.e., high-linear energy transfer radiation (see Chapter 6, Section A.4)] such as α particles, fission fragments, and other nuclear particles, are more damaging per gray of absorbed dose than is less densely ionizing radiation, such as β particles and γ rays.

The dose-modifying factors in this list are taken into account in preparing regulations and making recommendations for handling of radioactive materials. For example, regulations specify different dose limits for different parts of the body, for different time periods, and for different age groups. To account for the differing hazards of different types of radiation, the *equivalent dose*, defined previously (see Chapter 22, Section A), is used. For most of the radiation encountered in nuclear medicine, the equivalent dose in sieverts (or rems) is numerically equal to the absorbed dose in grays (or rads), although it must be emphasized that equivalent dose and absorbed dose are not the same quantity and have different units. To account for differing hazards for different organs and tissue types, the equivalent dose is modified by organ-specific weighting factors to compute the *effective dose* (see Chapter 22, Section B.7) to an individual.

In some older texts, and in current United States federal regulations (see Section B), the related quantities *dose equivalent* (in place of equivalent dose) and *effective dose equivalent* (in place of effective dose) may be encountered. The conceptual difference is that equivalent dose is based on the average absorbed dose in a specific tissue or an organ, whereas dose equivalent is based on the absorbed dose at a point in tissue. There also are differences in the scaling factors used to convert the absorbed dose into these quantities. These quantities are summarized in Table 23-1. Broadly speaking, for nuclear medicine applications, equivalent dose and dose equivalent, as well as effective dose and effective dose equivalent, have similar numerical values.

2. Exposure and Air Kerma

For the purpose of describing radiation *levels* in a radiation environment, an additional quantity—*exposure*—has traditionally been used. Exposure refers to the amount of

TABLE 23-1
QUANTITIES USED IN HEALTH PHYSICS

Quantity	Symbol	Units	Definition	Comment
Equivalent dose	H_T	Sv	Average absorbed dose across a tissue or organ T with weighting factors that depend on the type and energy of radiation. See Chapter 22, Section A.	Replaces dose equivalent. See ICRP Publication 60 and updated radiation weighting factors in ICRP Publication 103.
Effective dose	E	Sv	Measure of absorbed dose to whole body based on multiplying equivalent dose by organ-specific weighting factors. See Chapter 22, Section B.7.	See ICRP Publication 60 (1991) and updated tissue weighting factors in ICRP Publication 103 (2007).
Dose equivalent	H	Sv	Absorbed dose at a point in an organ, with quality factors that depend on the type of radiation. See ICRP Publication 51.	Replaced by equivalent dose in ICRP Publication 60 but still used in U.S. Federal regulations in 2012.
Effective dose equivalent	H_E	Sv	Introduced in ICRP Publication 26 (1977) as a measure of effective radiation dose to the whole body. Is based on dose equivalent values multiplied by tissue weighting factors.	Replaced by effective dose in ICRP Publication 60 (1991) but still used in U.S. Federal regulations in 2012.
Exposure	X	C/kg	Amount of charge liberated per kg of air by a γ-ray or x-ray source.	Traditional units were the Roentgen (R) in which 1R = 2.58 × 10^{-4} C/kg. Exposure replaced by air kerma.
Air kerma	K	Gy	Amount of kinetic energy released per kg of air by uncharged ionizing radiation (photons and neutrons).	For radionuclides used in nuclear medicine, the conversion between air kerma and exposure is K(Gy) ≈ X(C/kg) × 33.7.

References:
ICRP Publication 26: *Ann ICRP* 1: 3, 1977.
ICRP Publication 51: *Ann ICRP* 17: 2-3, 1987.
ICRP Publication 60: *Ann ICRP* 21: 1-3, 1991.
ICRP Publication 103: *Ann ICRP* 37: 2-4, 2007.

ICRP, International Commission on Radiological Protection.

ionization of air caused by a γ-ray or x-ray source. The traditional unit of exposure is the *roentgen* (R), with subunits of milliroentgens (1 mR = 10^{-3} R), microroentgens (1 μR = 10^{-6} R), and so on. An exposure of 1 R implies ionization liberating an amount of charge equal to 2.58 × 10^{-4} coulombs/kg of air, or approximately 2 × 10^9 ionizations per cc of dry air at standard temperature and pressure. An *exposure rate* of 1 R/min implies that this amount of ionization is produced during 1 minute. The SI unit for exposure is the coulomb/kg, with no special name. Thus 1 coulomb/kg ≈ 3876 R and 1 R = 2.58 × 10^{-4} coulombs/kg.

The use of the SI units for exposure is cumbersome, and therefore in the transition to SI units, exposure is being replaced by a related quantity known as *air kerma*. *Kerma* stands for *k*inetic *e*nergy *r*eleased in *m*edi*a*. *Exposure* refers to the ionization charge produced in air, whereas *air kerma* refers to the amount of kinetic energy released in air (Table 23-1). More precisely, the air kerma is the sum of the kinetic energy of all charged particles produced by interactions from a source of x rays or γ rays (through Compton scatter, photoelectric absorption, or pair production) per kg of air. The units of air kerma are grays (J/kg), the same as for absorbed dose. If all of the photon energy transferred to charged particles is deposited locally (in air, bremsstrahlung production is negligible, so this is a reasonable assumption), then the absorbed dose in air has the same value as the air kerma. Using the fact that 33.7 eV of energy

is required to produce an ion pair in air (see Table 7-1), and assuming bremsstrahlung losses can be ignored, the relationship between exposure, X, and air kerma, K, can be calculated as:

$$K(\text{Gy}) \approx X(\text{C/kg}) \times 33.7 \qquad (23\text{-}1)$$

The conversion between traditional units of exposure and air kerma is given by:

$$K(\text{Gy}) \approx X(\text{R}) \times 0.00869 \qquad (23\text{-}2)$$

Exposure and air kerma are useful quantities because they can be measured using ionization chambers, which are basically ionization-measurement devices (Chapter 7, Section A.2). Specific instruments used for health physics measurements are described in Section E.

If the air kerma in Gy is known at a certain location, the absorbed dose in Gy that would be delivered to a person at that location can be estimated by means of a scaling factor, f. This factor is defined as the ratio of the absorbed dose in the medium of interest, D_{med}, to the absorbed dose in air, D_{air}:

$$f = D_{\text{med}}/D_{\text{air}} \approx D_{\text{med}}/K \qquad (23\text{-}3)$$

The factor f depends on the mass attenuation coefficients (Chapter 6, Section D.1) of the medium of interest and of air and is energy dependent. Figure 23-1 shows the value of f as a function of energy for bone and for soft tissues. For soft tissues, $f \approx 1.1$. The value is close to unity because the mass attenuation properties of soft tissues and air are similar. For low-energy photons ($E \lesssim 100$ keV), the value of f for bone is greater than unity. Because of photoelectric absorption by the heavier elements in bone (Ca and P), energy absorption in bone is greater than energy absorption by air at these energies; however, for most of the γ-ray energies commonly employed in nuclear medicine, the value of f for bone also is close to 1.

Thus for practical purposes air kerma (in grays) is approximately equal numerically to the absorbed dose in grays that would be received by an individual at that location, and in turn, as described in Section A.1, the absorbed dose in grays is numerically equal to the equivalent dose in sieverts. Because of their approximate numerical equivalence, grays and sieverts, or in traditional units, roentgens, rads and rems, are sometimes (mis)used as approximately interchangeable quantities; however, one should be aware that

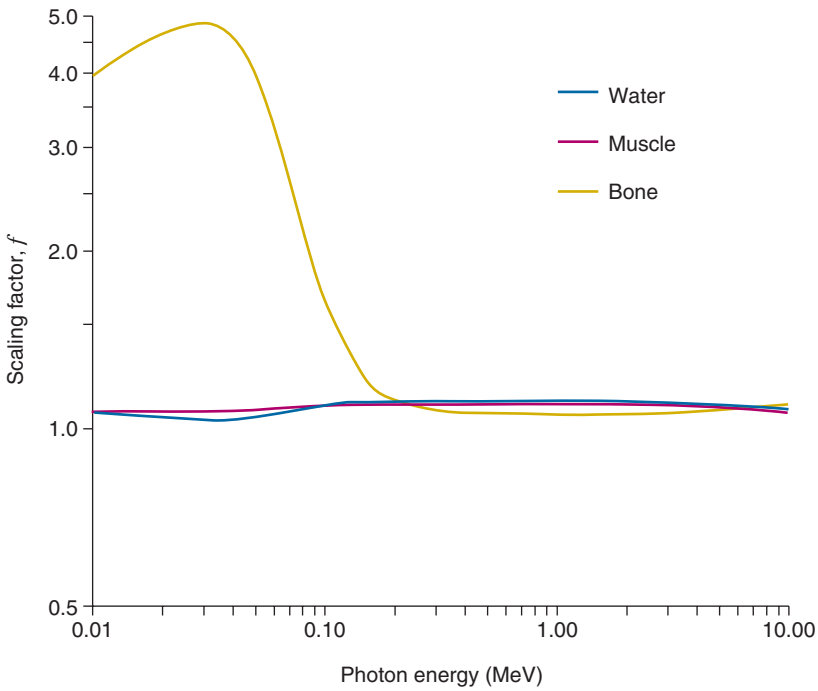

FIGURE 23-1 Scaling factor f versus photon energy for water, muscle, and bone.

they represent distinctly different physical quantities.

B. REGULATIONS PERTAINING TO THE USE OF RADIONUCLIDES

Regulations for the transport, handling, and exposure to ionizing radiation vary from country to country. The discussion in this section limits itself to the regulations in place in the United States in 2012 and uses selected regulations to highlight important regulatory concepts. A complete discussion of the many regulations involved is beyond the scope of this chapter. Also, the regulations are under constant review and subject to periodic changes. Therefore the regulations presented in this section should not be used to determine compliance without checking that they are still current. Further information may be obtained in the references at the end of the chapter or from institutional health physicists.

1. Nuclear Regulatory Commission Licensing and Regulations

The use and distribution of radioactive materials in the United States are under the primary control of the Nuclear Regulatory Commission (NRC). The NRC issues licenses to individuals and to institutions to possess and use radioactive materials. In addition to medical uses, industrial, research, educational, and other uses of radioactive materials also require NRC licensing. In some states, the NRC has entered into an agreement to transfer its regulatory and licensing functions to a radiation control agency within the state. Such states are called *agreement states.*

Medical licenses generally fall into one of two categories: specific licenses of limited scope or specific licenses of broad scope. Limited-scope licenses are for limited kinds and quantities of radionuclides, which are listed specifically in the license. They may be issued to individual physicians (e.g., in private offices) or to institutions (e.g., hospitals). Licenses issued to institutions also list the of individuals authorized to practice under the license.

Broad-scope licenses are issued to larger institutions that require greater licensing flexibility (e.g., basic research as well as medical uses in a university setting). Broad-scope licenses generally cover more radionuclides and greater quantities than do limited-scope licenses. The NRC permits the institutional radiation safety committee to authorize individuals to use radionuclides under the license rather than requiring them to be listed specifically on the license.

The NRC also issues regulations that must be observed by licensees in the use of radioactive materials. These regulations are published in Title 10 of the Code of Federal Regulations (CFR). Two of the more relevant sections of these regulations for nuclear medicine are Part 20 (10CFR20), covering radiation protection, and Part 35 (10CFR35), covering medical uses. The NRC regulations are based primarily on the recommendations of two advisory bodies, the ICRP and the National Council on Radiation Protection and Measurement (NCRP), as discussed in Section B.7. The NRC also periodically issues regulatory guides to assist licensees in the interpretation and implementation of its regulations.

In addition to the NRC, several other government agencies are involved in the regulation of radioactive materials, such as the U.S. Department of Transportation (shipping regulations) and the U.S. Food and Drug Administration (pharmaceutical aspects).

2. Restricted and Unrestricted Areas

The NRC regulations prescribe different maximum radiation limits for restricted and unrestricted areas. A restricted area is one "… access to which is controlled by the licensee for the purposes of protection of individuals from exposure to radiation and radioactive materials." Normally, restricted areas are not accessible to the general public, and generally they are occupied only by individuals whose employment responsibilities require them to work with radioactive materials and other radiation sources. Such individuals (e.g., nuclear medicine physicians, technicians, and radiochemists) are said to be *occupationally exposed.* Administrative staff, janitorial personnel, and facilities maintenance personnel generally are not included in this category. Restricted areas must be clearly marked with radiation warning signs.

3. Dose Limits

The *dose limits* specified in 10CFR20 are based on the general recommendations by the ICRP and NCRP (Section B.7) that an individual's total effective dose (see Chapter 22, Section B.7) should not exceed 50 mSv (5 rem) per year. Furthermore, 10CFR20 requires that the deep-dose equivalent (dose

equivalent at a depth of 1 cm in tissue) to any individual organ or tissue (excluding the lens of the eye) should not exceed 500 mSv (50 rem) per year. The limit for shallow-dose equivalent (dose equivalent at a depth of 0.007 cm in tissue) to the skin and extremities also is 500 mSv (50 rem) per year. The most restrictive limit is to the lens of the eye, which has an annual limit of 150 mSv (15 rem). The annual occupational dose limits for minors (<18 years of age) are 10% of the annual dose limits specified for adult workers. The dose equivalent to an embryo or fetus should not exceed 5 mSv (0.5 rem).

These dose limits, which apply to occupationally exposed personnel, are called *occupational dose limits*. Occupational dose limits do not include radiation doses received by the occupationally exposed individual while that individual is undergoing a medical examination, nor do they include any radiation dose from natural radiation sources, such as cosmic rays and naturally occurring radioactivity in the environment.

Note that the regulations require the licensee to control radiation doses not only from licensed materials but from "other sources in the licensee's possession" as well (e.g., nonlicensed radioactive materials or an x-ray generator). Thus a licensee would be in violation of the regulations if the occupation limits were exceeded even if most of the radiation dose were caused by nonlicensed sources.

For individual members of the public, the annual effective dose equivalent limits are 1 mSv (0.1 rem). Radiation levels in *unrestricted areas* should deliver a radiation dose of less than 0.5 μSv/hr (0.05 mrem/hr), assuming continuous occupation of the area. Transient radiation levels of up to 20 μSv/hr (2 mrem/hr) are permitted.

4. Concentrations for Airborne Radioactivity in Restricted Areas

A particular problem in nuclear medicine laboratories is the potential for leakage or escape of radioactive gases (e.g., ^{133}Xe used in pulmonary function studies) or volatile radioactive material (e.g., concentrated ^{131}I solutions). The NRC regulations specify the concentrations for airborne radioactive materials that would result in the annual dose limits described in Section 3. These calculations assume that the workers are chronically exposed to these concentrations during a 2000-hour working year and that 2×10^4 mL of air is breathed per minute. Concentration limits are shown in Table 23-2 for radionuclides that are used in nuclear medicine.

5. Environmental Concentrations and Concentrations for Sewage Disposal

The NRC regulations also specify the environmental concentrations of radioactivity in air and water and the concentration of radionuclides disposed of into sewage water, which would lead to the annual dose limits described in Section B.3 for the general public. Radioactive concentrations in sewage are of concern because they may eventually reach public water supplies. These limits assume continuous inhalation or ingestion by the general public over a period of 1 year, and they further assume that the average person breathes 2×10^4 mL of air per minute and has an annual water intake of 7.3×10^5 mL. Sewer water is assumed to be diluted by a factor of 10 before it is ingested. The methods for calculating these concentrations are described in 10CFR20. Table 23-3 shows these concentrations for several radionuclides of interest.

6. Record-Keeping Requirements

The NRC regulations require that rather extensive records be kept by the licensee. These include, among others, personnel dosimetry and radiation survey records (Section E), wipe testing records for sealed sources, summaries of quality control checks on radiation monitoring equipment, inventory

TABLE 23-2
CONCENTRATION OF AIRBORNE RADIOACTIVITY THAT WOULD RESULT IN THE ANNUAL DOSE LIMITS DESCRIBED IN SECTION B.3 FOR OCCUPATIONALLY EXPOSED PERSONNEL

	Air Concentration	
Radionuclide	μCi/mL	kBq/mL
^{3}H	2×10^{-5}	0.74
^{11}C	2×10^{-4}	7.4
^{14}C	1×10^{-6}	3.7×10^{-2}
^{18}F	3×10^{-5}	1.1
^{99m}Tc	6×10^{-5}	2.2
^{125}I	3×10^{-8}	1.1×10^{-3}
^{131}I	2×10^{-8}	7.4×10^{-4}
^{133}Xe	1×10^{-4}	3.7

Data from 10CFR20, Appendix B, Table 1.

TABLE 23-3
ENVIRONMENTAL CONCENTRATIONS (AIRBORNE AND WATER) AND SEWAGE CONCENTRATIONS THAT WOULD RESULT IN THE ANNUAL DOSE LIMITS DESCRIBED IN SECTION B.3 FOR THE GENERAL PUBLIC IF CONTINUOUSLY INHALED OR INGESTED

	Environmental Concentrations					
	Air		Water		Sewage Concentration	
Radionuclide	μCi/mL	kBq/mL	μCi/mL	kBq/mL	μCi/mL	kBq/mL
^{3}H	1×10^{-7}	3.7×10^{-3}	1×10^{-3}	3.70×10^{1}	1×10^{-2}	3.70×10^{2}
^{11}C	6×10^{-7}	2.2×10^{-2}	6×10^{-3}	2.22×10^{2}	6×10^{-2}	2.22×10^{3}
^{14}C	3×10^{-9}	1.1×10^{-4}	3×10^{-5}	1.11×10^{0}	3×10^{-4}	1.11×10^{1}
^{18}F	1×10^{-7}	3.7×10^{-3}	7×10^{-4}	2.59×10^{1}	7×10^{-3}	2.59×10^{2}
^{99m}Tc	2×10^{-7}	7.4×10^{-3}	1×10^{-3}	3.70×10^{1}	1×10^{-2}	3.70×10^{2}
^{125}I	3×10^{-10}	1.1×10^{-5}	2×10^{-6}	7.40×10^{-2}	2×10^{-5}	7.40×10^{-1}
^{131}I	2×10^{-10}	7.4×10^{-6}	1×10^{-6}	3.70×10^{-2}	1×10^{-5}	3.70×10^{-1}
^{133}Xe	5×10^{-7}	1.9×10^{-2}	—	—	—	—

Data from 10CFR20, Appendix B, Tables 2 and 3.

and disposal records, minutes of radiation safety committee meetings, and records of training in radiation safety of laboratory personnel. Maintenance of proper records is one of the major activities of an NRC licensee.

7. Recommendations of Advisory Bodies

The NRC regulatory limits are based on recommended radiation dose limits published by various advisory bodies. These bodies include the NCRP, a U.S. organization, and two international groups, the ICRP and the International Commission on Radiological Units (ICRU). The last group is concerned mostly with definitions of radiologic units. Recommendations from these groups do not carry the force of law; however, there is a tendency of regulatory agencies such as the NRC to convert them into law. Therefore, it is worthwhile to keep abreast of their recommendations.

Some titles of NCRP and ICRP reports that are applicable to nuclear medicine are listed in the references at the end of the chapter. Table 23-4 lists the dose limits currently recommended by the NCRP and the ICRP. Note that their coverage is somewhat broader than those appearing in the NRC regulations. Note also the restrictive limits placed on pregnant women with respect to the fetus. A pregnant, occupationally exposed woman may require special work restrictions to ensure that this dose limit for the fetus is not exceeded during her pregnancy.

C. SAFE HANDLING OF RADIOACTIVE MATERIALS

1. The ALARA Concept

Radiation dose limits and other restrictions specified in NRC regulations are legal limits that must not be exceeded at any time by an NRC licensee; however, they should not be considered as thresholds below which exposure to radiation is of no concern. Presently, although the radiation hazards associated with the limits specified in the regulations are very small, they are not assumed to be totally risk free, and any reasonable technique for reducing radiation dose may have potential benefits in the long run.

Recognizing this, the NCRP as early as 1954, and more recently the NRC regulations, have recommended as an operating philosophy that the objective of radiation safety practices should be not simply to keep radiation doses within legal limits but to keep them "*a*s *l*ow *a*s *r*easonably *a*chievable" (ALARA). In its regulations, the NRC has defined ALARA to mean "as low as reasonably achievable taking into account the state of technology and economics of improvement in relation to benefits to the public health and safety, and other societal and socioeconomic considerations, and in relation to the use of atomic energy in the public interest." NRC Regulatory Guides 8.10 ("Operating Philosophy for Maintaining Occupational Exposures as Low

TABLE 23-4
DOSE LIMITS RECOMMENDED BY THE INTERNATIONAL COMMISSION ON RADIOLOGICAL PROTECTION AND THE NATIONAL COUNCIL ON RADIATION PROTECTION AND MEASUREMENT

	NCRP* (1993)	ICRP (2007)†
Occupational Exposure		
Effective dose: Annual	50 mSv	20 mSv/year averaged over 5 years, no more than 50 mSv in any one year
Effective dose: Cumulative	10 mSv × age (years)	100 mSv in any 5-year period
Equivalent dose: Annual	150 mSv to lens of eye 500 mSv to skin, hands, and feet	150 mSv to lens of eye 500 mSv to skin, hands, and feet
General Public Exposure		
Effective dose: Annual	1 mSv if continuous 5 mSv if infrequent	1 mSv; higher if needed as long as average over 5 years does not exceed 1 mSv
Equivalent dose: Annual	50 mSv to skin, hands, and feet	15 mSv to lens of eye 50 mSv to skin
Embryo-Fetus		
Equivalent dose	0.5 mSv per month once pregnancy declared	Same as general public exposure, even for occupational exposure

ICRP, International Commission on Radiological Protection; NCRP, National Council on Radiation Protection and Measurement.

*Limitation of Exposure to Ionizing Radiation (NCRP Report No. 116), 1993.

†2007 Recommendations of the International Commission on Radiological Protection (ICRP Publication No. 103). Annals of the ICRP, 2-4, 2007.

as Reasonably Achievable") and 8.18 ("Information Relevant to Ensuring that Occupational Radiation Exposures at Medical Institutions will be as Low as Reasonably Achievable") provide practical advice on implementation of ALARA principles.

The concept of ALARA has long been the operational objective of radiation safety practices in well-run nuclear medicine laboratories, and they have now taken on regulatory force. ALARA principles can be applied to the handling of radiation sources, to storage and shielding techniques, and to the design and layout of the laboratory. Some of the basic techniques for keeping radiation doses "ALARA" are discussed later.

2. Reduction of Radiation Doses from External Sources

Types of Sources. External sources are those that deliver a radiation dose from outside the body. The principal sources are γ-ray- and x-ray-emitting radionuclides in patients, syringes, vials, waste disposal areas, and so forth. Unshielded emitters emitting particles of sufficient energy to travel some distance in air (e.g., ^{32}P, but not ^{14}C) also constitute an external hazard, although β particles generally deliver only a superficial radiation dose to the skin.

Air Kerma Rate Constant. The air kerma caused by γ-ray and x-ray emitters can be estimated from the air kerma rate constant, Γ. This constant has a specific value for each radionuclide and is defined as the air kerma caused by γ-ray and x-ray emissions, in mGy per hour, at a distance of 1 m from an unshielded 1-GBq point source of that radionuclide. The units for Γ are mGy • m^2/Gbq • hr. Calculation of the air kerma rate constant is based on the number of γ-ray and x-ray emissions from the radionuclide (number per disintegration) and their energies, and on the absorption coefficient of air at these energies. Values of Γ for some radionuclides used in nuclear medicine are summarized in Table 23-5. For practical health physics purposes, the calculation of Γ should only include γ and x rays above a certain minimum energy value because photons of a lower energy have such low penetrating power (e.g., through the walls of a syringe or vial) that they pose a negligible external hazard. The minimum energy used to compute the values in Table 23-5 is 20 keV.

To estimate the air kerma rate KR(mGy/hr) at a distance d(m) from an activity A(GBq) of a radionuclide having an air kerma rate constant Γ(mGy • m^2/GBq • hr), the following equation is used:

$$KR = A\Gamma/d^2 \qquad (23\text{-}4)$$

TABLE 23-5
γ RAY AIR KERMA RATE CONSTANTS FOR SEVERAL RADIONUCLIDES OF INTEREST IN NUCLEAR MEDICINE

Radionuclide	Γ (mGy · m²/GBq · hr)	Radionuclide	Γ (mGy · m²/GBq · hr)
^{11}C	0.1393	^{99m}Tc	0.0141
^{13}N	0.1394	^{111}In	0.0831
^{15}O	0.1395	^{123}I	0.0361
^{18}F	0.1351	^{125}I	0.0377
^{57}Co	0.0141	^{131}I	0.0522
^{60}Co	0.3090	^{133}Xe	0.0143
^{67}Ga	0.0195	^{137}Cs/^{137}Ba	0.0821
^{68}Ga	0.1290	^{201}Tl	0.0102
^{99}Mo/^{99m}Tc (at equilibrium)	0.0336		

Data from Ninkovic MM, Raicevic JJ, Adrovic F: Air kerma rate constants for gamma emitters used most often in practice. *Radiat Prot Dos* 115:247-250, 2005.

The appearance of d^2 in the denominator of Equation 23-4 is an expression of the *inverse-square law* (see Chapter 11, Section A.2). Because γ rays and x rays are emitted isotropically (e.g., with no preferred direction) radiation intensity and dose levels decrease as the square of the distance from the source.

EXAMPLE 23-1

Calculate the air kerma rates at 10-cm and 300-cm distances from a syringe containing 1 GBq of ^{99m}Tc.

Answer

The air kerma rate constant Γ is 0.0141 mGy • m²/GBq • hr (see Table 23-5). Therefore from Equation 23-4, at 10 cm

$$KR = 1\ \text{GBq} \times 0.0141\ \text{mGy} \cdot \text{m}^2/\text{GBq} \cdot \text{hr} \div (0.1^2)\ \text{m}^2 = 1.41\ \text{mGy/hr}$$

and at 300 cm,

$$KR = 1\ \text{GBq} \times 0.0141\ \text{mGy} \cdot \text{m}^2/\text{GBq} \cdot \text{hr} \div (3^2)\ \text{m}^2 = 1.57 \times 10^{-3}\ \text{mGy/hr} = 1.57\ \mu\text{Gy/hr}$$

The strong effect of distance on radiation dose equivalent rate is illustrated by Example 23-1.

In some texts, exposure rates, *XR* (roentgens/hr), may be encountered instead of air kerma rates. The relationship between the two is given by

$$XR\ (\text{R/hr}) \approx KR\ (\text{mGy/hr}) \times 0.115 \quad (23\text{-}5)$$

The equivalent dose rate to a particular tissue or organ and the effective dose rate to an individual can be estimated from the air kerma rates using the appropriate radiation weighting factors and tissue weighting factors (see Chapter 22, Sections A and B.7), but strongly depend on patient geometry and the direction from which the radiation is incident on the individual. Some publications estimate dose equivalent rates in mSv/hr using simplified tissue models. These values are numerically higher than the air kerma rates as they include the conversion from air kerma to tissue absorbed dose (a factor of ~1.11 in soft tissue), and also are increased by the contributions of Compton-scattered photons within the body.

Equation 23-4 is accurate for distances that are large in comparison to the physical size of the source; however, it is not valid at very small distances. For example, it predicts that the air kerma rate, and therefore the equivalent dose, becomes infinite as d approaches zero. A practical situation in which this problem arises is in the estimation of equivalent dose rates on contact with the source, for example, equivalent dose rates to the hand while handling syringes and vials. These equivalent dose rates have been determined experimentally for ^{99m}Tc. They range

from about 0.14 mSv/MBq • hr on the surface of larger syringes (10 to 20 mL) up to approximately 0.7 mSv/MBq • hr for smaller syringes (1 to 2 mL).[7] Equivalent dose rates to the hands on contact with syringes containing ~1000 MBq of ^{99m}Tc can be in the range of several mSv/min, an obvious matter of concern in operations requiring handling of these sources. The use of syringe shielding therefore is indicated and can significantly reduce the radiation dose.

Time, Distance, Shielding (*TDS*) *Rules.* The basic principles for reducing radiation doses from external sources are described by the "TDS" rules, for time, distance, and shielding:

1. Decrease the *time* of exposure.
2. Increase the *distance* from the source.
3. Use *shielding* when practical and effective.

Time of exposure is decreased by working with or in the vicinity of radiation sources as rapidly as possible, consistent with good technique. Personnel should spend as little time as possible in "hot labs" and other high-level radiation areas. In particular, these areas should not be used for visiting, discussing problems unrelated to activities in the area, and so on. Laboratory monitors should be used in these areas to warn personnel when high-level radiation sources are present.

As shown by Example 23-1, *distance* can have a marked effect on radiation levels. Increasing distance always has a dose-reduction effect. Direct contact with radiation sources should be avoided by any available means, such as by using tongs to handle vials. Patient study areas (e.g., imaging rooms) should be arranged to permit the technician to operate instrumentation at reasonable distances (e.g., $\gtrsim$2 m) from the patient. Separate waiting areas should be provided for patients who have been injected with radioactivity and for relatives, orderlies, and patients not requiring radioactive injections. Reception areas should not be used as waiting areas for radioactive patients. Storage areas for generators, radioactive trash, and other high-level sources should be remote from regularly occupied areas of the laboratory. Special attention should be given to their location in relation to unrestricted areas. (They also should be remote from imaging rooms and counting rooms to minimize instrument background levels.)

Examples of effective use of *shielding* are lead pigs for storage of vials and generators, lead-lined syringe holders, lead aprons, lead bricks for lining storage areas, and lead-lined drawing stations (Fig. 23-2). Leaded glass provides comfortable viewing and radiation protection simultaneously, especially

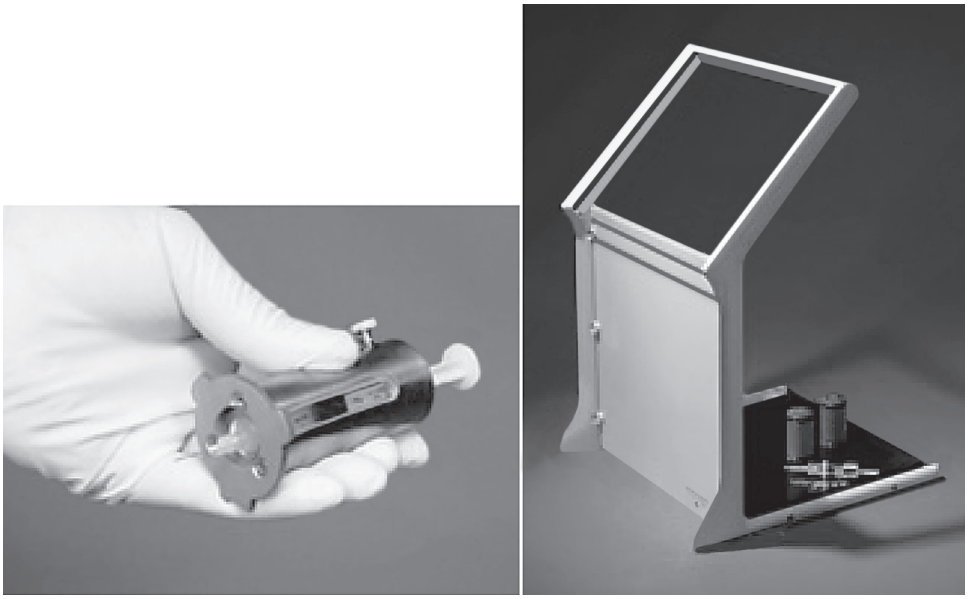

FIGURE 23-2 Examples of protective shielding devices used in nuclear medicine. *Left,* Shielded syringe holder designed for positron-emitting radionuclides. *Right,* Lead-lined drawing station for preparing and handling low-energy gamma-emitting radioactive materials ($\lesssim$150 keV) such as ^{99m}Tc. Lead-lined glass provides a good view of the work area. (*Photographs courtesy Biodex Medical Systems, Inc., Shirley, NJ.*)

for low-energy γ-ray and x-ray emitters (≲200 keV). Dose calibrators should be enclosed in a shielded area, using lead sheets or bricks, to avoid unnecessary exposure during measurement of radiopharmaceutical activity.

Table 6-4 lists tenth-value thicknesses of lead for several γ-ray and x-ray emitters. Small thicknesses of lead (≲1 mm) provide effective shielding for low-energy emitters (e.g., ^{133}Xe and ^{99m}Tc). Lead-lined aprons, which usually contain 0.25-mm- or 0.5-mm -equivalent lead thicknesses, provide a modest amount of radiation protection, but probably not enough to warrant their routine use in the nuclear medicine laboratory; however, they may be useful for specific applications, such as during handling of large quantities of ^{133}Xe (E_γ = 81 keV) or during elution of a ^{99m}Tc generator (E_γ = 140 keV). Greater thickness (>1 cm) is required for higher-energy γ-ray emitters, such as ^{131}I (E_γ = 364 keV); however, lead is still an effective shielding material at these energies. Concrete and similar materials find limited use for general purpose shielding in nuclear medicine.

Shielding is very effective for β emitters. A few millimeters of almost any solid material will stop even the most energetic β particles (see Fig. 6-10 and Table 6-1). In this case, however, low-Z materials (e.g., plastic, ordinary glass) are preferred over high-Z materials (e.g., leaded glass) to minimize bremsstrahlung production (see Equation 6-1). A good shielding arrangement for a high-energy β emitter, such as ^{32}P, is to use a plastic or glass container for the radioactive material to stop the β particles and then to place this in a lead container to absorb the bremsstrahlung radiation (see Fig. 6-3). A similar approach can be employed with positron emitters; however, the thickness of the lead must be substantial (tenth-value thickness for 511-keV photons is 13.4 mm) to provide effective shielding against the annihilation radiation.

3. Reduction of Radiation Doses from Internal Sources

Types of Sources. Nearly all nuclear medicine personnel are required at one time or another to work with radioactive sources in open or poorly sealed containers. There is always the possibility that in these operations some of the radioactive material will find its way into the body, where it delivers a radiation dose as an internal radiation source, or back to offices or other areas accessible to nonradiation workers. Patient sweat or excreta, linens used on imaging tables, spillage occurring during transfer of activity between containers and syringes, radioactive trash, and radioactive gases released during pulmonary function tests are examples of potential sources.

A radioactive material that has been accidentally or carelessly ingested is an "uncontrolled source"; once it is inside the body, there is very little that can be done to reduce the radiation dose that it will deliver. (Techniques developed by the weapons and reactor industries for speeding the elimination of radioactive materials from the body generally are slow to act and thus are useful only for very long-lived radionuclides, and impractical for nuclear medicine.) The cardinal rule for keeping radiation doses from internal sources ALARA is to prevent the entry of the radioactive material into the body in the first place. To a certain extent, this is a matter of careful design of laboratory facilities, but equally as important, it is a matter of developing good laboratory work habits.

Some basic rules for avoiding internal radiation doses are the following:

1. No eating, drinking, smoking, or applying of cosmetics should occur in areas where open sources may be present (e.g., hot labs and patient study areas).
2. Lab coats and gloves should be worn when handling radioactive sources. Gloves should be handled so as to avoid contamination of their inside surfaces. Lab coats, aprons, and other protective clothing should stay in the laboratory (i.e., they should not worn outside the lab or taken home).
3. No foodstuffs or drinks should be stored where radioactive sources are kept, such as in laboratory refrigerators.
4. Pipetting should never be done by mouth.
5. Personnel should wash their hands after working with radioactive sources (a sink should be available in the laboratory), and they should be checked for contamination on a laboratory radiation monitor (Section E.1). Hands should also be monitored before going to lunch or on breaks and before leaving at the end of the day.
6. Work should be performed on absorbent pads to catch spills and prevent spattering of liquids.
7. Work with radioactive gases or other volatile materials (e.g., concentrated iodine solutions) should be performed

in a ventilated fume hood. These materials also should be stored in a hood.
8. Work areas should be kept tidy. Radioactive trash, contaminated pads, and so forth should be disposed of promptly.
9. Radioactive storage areas (e.g., hot labs) should not be used to store other materials, such as office supplies or linens.
10. Needless contamination of light switches, doorknobs, and other items that could result in unsuspected contamination to personnel should be avoided.
11. Containers with sharp or broken edges should not be used for radioactive materials.
12. Radioactive materials should be stored when they are not in use.

Studies with radioactive gases, such as ^{133}Xe, require special attention because of the potential for escape of radioactivity into the laboratory and beyond. Optimally, the laboratory ventilation system should be designed to maintain the laboratory under negative pressure relative to its surroundings and should be separate from other ventilation systems to prevent spread of airborne activity into other areas of the hospital. A gas-trapping system should be used to collect gases exhaled by the patient.

Most of the rules listed earlier in this section are of the common-sense variety and perhaps seem obvious; however, it is surprising how often they are violated through forgetfulness or indifference. This may explain the surprisingly high incidence of internal radionuclides found in some studies of nuclear medicine laboratory personnel (e.g., >70% incidence of radioactive iodine in thyroid glands).[6] Clearly, adherence to proper laboratory work rules is fundamental to the ALARA concept.

4. Laboratory Design

The principles of ALARA are enhanced by careful attention to laboratory design. Several design aspects have been mentioned already in relation to other problems, such as negative relative air pressure in laboratories employing volatile or gaseous radioactive materials and availability of a fume hood with its own exhaust system for storage of these materials. Some additional principles to be considered in laboratory design are the following:

1. Hot labs and radioactive storage areas should be located away from other busy work areas, public corridors, secretarial offices, and so on, and away from imaging and low-level counting rooms.
2. Work surfaces and floors should be constructed using smooth, nonabsorbent materials free from cracks and crevices.
3. Workbenches should be sufficiently sturdy to support lead shielding.
4. Washbasins and sinks should be conveniently available where unsealed radioactive materials are handled. It is desirable that sinks in hot labs have foot- or elbow-operated controls.
5. The laboratory design should permit separate storage of glassware and work tools (e.g., tongs, stirring devices) not used with radioactive materials to prevent needless contamination or mixture with similar items used with radioactive preparations.

5. Procedures for Handling Spills

Accidental spills of radioactive materials are infrequent occurrences in well-run nuclear medicine laboratories. Also, the quantities of radioactivity used in nuclear medicine do not create "life-threatening" hazards. Nevertheless, radioactive spills should not be treated as events completely without hazard, and laboratory personnel should be aware of the appropriate procedures to follow when spills do occur.

The steps to follow in dealing with a radioactive spill are (1) to *inform,* (2) to *contain,* and (3) to *decontaminate.*

1. Individuals in the immediate work area should be informed that a spill has occurred so they can avoid contamination if possible. Individuals outside the immediate area should be warned so they do not enter it. The radiation officer should be informed so that he or she may begin supervising further action as soon as possible.
2. By whatever means are reasonably possible, but without risking further hazards to themselves, laboratory personnel should attempt to contain the spill to prevent further spread of contamination. A flask that has been tipped over should be uprighted. Absorbent pads should be thrown over a liquid spill. Doors should be closed to prevent the escape of airborne radioactivity (e.g., gases, powders). The spill area should be closed off to prevent entry, especially by persons who might not be aware of the spill. Personnel monitoring for contamination should be started as soon as possible, so that

contaminated and uncontaminated persons can be segregated. To prevent the further spread of radioactivity, contaminated individuals should not be allowed to leave the area until they are decontaminated, and uncontaminated individuals (with the exception of appropriately protected emergency personnel and other designated personnel involved with the cleanup) should not be allowed to enter the spill area. Contamination monitoring should be done using a sensitive radiation monitoring instrument appropriate for the type of radioactivity involved. It is advisable that each laboratory have on hand a thin-window Geiger-Müller (GM) counter survey meter (Section E.1) for handling such situations.

3. Personnel decontamination procedures should receive first priority, followed by decontamination of work areas, and so on. Personnel involved in decontamination procedures should wear protective clothing to avoid becoming contaminated themselves in the process. Contaminated skin should be flushed thoroughly with water. Special attention should be given to open wounds and contamination around the eyes, nose, and mouth. Contaminated clothing should be removed and placed in plastic bags for storage. After major localized areas of personnel contamination have been attended to, a shower bath may be required to remove more widely distributed contamination.

Decontamination of laboratory and work areas should not be attempted except under the supervision of the radiation safety officer or radiation health physicist. If the work surfaces and floors are constructed from a nonabsorbent material, soap and water is generally all that is needed for decontamination. Contaminated areas should be cleaned "from outside in" to minimize the spread of contamination. Porous or cracked surfaces may create difficult problems. If complete decontamination is not possible, it may be necessary to cover and shield the affected surfaces or perhaps even to remove and replace them.

D. DISPOSAL OF RADIOACTIVE WASTE

There are three general techniques for disposing of radioactive wastes.

1. *Dilute and disperse.* Small quantities of radioactive materials may be released into the environment—for example, radioactive gases into the ventilation system or liquid wastes into the sink—provided that the concentrations do not exceed the values specified in 10CFR20 (see Table 23-3). In keeping with the ALARA concept, however, this technique should not be used if reasonable alternatives are available (e.g., steps 2 and 3).
2. *Store and decay.* For materials having reasonably short half-lives (e.g., a few weeks or less), and if suitable storage space is available, this may be an economical and effective disposal technique. After a decay period of 10 half-lives has elapsed, only 0.1% of the initial activity remains. It is advisable to separate waste materials into two categories: those having half-lives shorter than 3 days and those having half-lives longer than 3 days, so that long-term accumulation of large volumes of waste material can be avoided. Disposal by decay of materials with half-lives longer than approximately 1 month is frequently impractical because of the long storage period required.
3. *Concentrate and bury.* This is frequently the only effective means of disposal of long-lived radioactivity, particularly if storage space is limited. A number of commercial companies provide this type of disposal service.

E. RADIATION MONITORING

1. Survey Meters and Laboratory Monitors

Survey meters are used to monitor radiation levels in and near laboratories where radioactive materials or other radiation sources are present. Generally, they are battery operated and portable. The radiation detector is usually an ionization chamber or a GM tube (see Figs. 7-3 and 7-11).

Ionization chamber types are calibrated to read exposure levels. The full-scale reading and range on the meter display is switch selectable, typically from 0 to 3 mR/hr up to 0 to 300 mR/hr. Many systems now also provide readings in units of air kerma. Some types have very thin mica or aluminum entrance windows for the ionization chamber and can be used to detect β particles as well as x rays or γ rays. Ionization chamber survey

meters give reasonably accurate estimates of exposure rates (±10%) over most of the nuclear medicine energy range. Most ionization chamber survey meters do not have sealed chambers. Thus for greatest accuracy their readings should be corrected for ambient temperature and pressure variations (see Chapter 7, Section A.2, Equation 7-1). These corrections are small at sea-level pressures and room temperatures; however, the pressure correction factor may be significant at higher elevations (~20% at 1600 m).

The accuracy of an ionization chamber survey meter should be checked periodically (e.g., annually, or following major repairs) using a radiation source producing a known radiation exposure level. Sealed sources used in radiation therapy departments are useful for this purpose.

GM tube types of survey meters are more sensitive than ionization chamber types because they respond to individual ionizing radiation events. Most of these instruments have meters that display event counting rates (cpm). Some types with thin mica or aluminum entrance windows are suitable for detecting α and β particles as well as γ rays and x rays. Because of their relatively high sensitivity, GM-type survey meters are most useful for detecting small quantities of radioactivity from minor spills, in waste receptacles, and so on.

Laboratory monitors are very similar to survey meters, but they are designed to be used at a fixed location rather than as portable units. They are operated continuously; thus they are generally plugged into the wall rather than battery operated. Most have GM tube detectors and produce an audible clicking noise when radiation is detected in addition to having a meter display of counting rate. A laboratory monitor should be used in any area where large quantities of radioactivity are handled (e.g., in a radiopharmacy laboratory) to warn of the presence of high radiation levels. They also are useful for monitoring hands after operations requiring the handling of radioactivity.

2. Personnel Dosimeters

Personnel dosimeters are devices worn by laboratory personnel to monitor radiation doses from external sources. There are two general types: *dosimeter badges,* which are used to measure cumulative doses over periods of weeks or months, and *pocket dosimeters,* which are generally used for monitoring over a shorter term.

Dosimeter badges monitor radiation doses using either a small piece of x-ray film, or much more commonly, thermoluminescent dosimeter (TLD) chips (Fig. 23-3). TLDs generally use small "chips" of LiF, a material that gives off light when heated after it has been exposed to ionizing radiation. The amount of light given off is measured using a photomultiplier tube while the chip is heated in an oven inside a light-tight enclosure. The amount of light given off is used to estimate the radiation dose received. There also are dosimeter badges based on optically stimulated (rather than heat-stimulated) luminescence of materials such as Al_2O_3.

Dosimeter badge services are provided by a number of commercial suppliers. New badges are supplied at regular (e.g., monthly) intervals, and readings for the preceding period are reported back to the user, typically

FIGURE 23-3 Examples of personnel dosimeter badges. *Left,* Thermoluminescent dosimeter (TLD) badge, which usually contains several TLD chips, with one chip being exposed through a thin Mylar window in the badge to permit measurement of low-energy beta radiation. *Right,* Photograph of ring dosimeters that contain a single lithium fluoride TLD chip and are useful for measuring the dose to the skin and hands from the handling of radionuclides. Bar codes are used on most personnel monitors to permit easy identification and data logging when reading out the TLD chips. (*Courtesy Mirion Technologies, Irvine, CA.*)

within about a month. The reports provided by most companies are satisfactory for NRC record-keeping purposes.

Pocket dosimeters were described in Chapter 7, Section A.2 (Fig. 7-5). They are essentially ionization chamber devices that provide an immediate readout of radiation doses and thus are especially useful for measuring over short periods or when a rapid indication of results is needed, such as during complicated radiopharmaceutical preparation procedures.

3. Wipe Testing

Wipe testing is used to detect small amounts of radioactive contamination on bench-top surfaces, on the outside of shipping packages, and so on, or to detect small amounts of radioactive leakage from sealed radioactive sources. The surface is wiped with an alcohol-soaked patch of gauze or cotton-tipped swab, which is then counted in a well counter (for γ-emitting nuclides) or a liquid scintillation counter (for β emitters). Contamination below the kBq level can be detected by wipe testing. NRC regulations require periodic wipe testing of work areas and maintaining records of these tests.

REFERENCES

1. Evaluation of the Linear Nonthreshold Dose-Response Model for Ionizing Radiation (NCRP Report No. 136). Bethesda, MD, NCRP, 2001.
2. Rossi HH: Sensible radiation protection. *Health Physics* 70:394-395, 1996.
3. Simmons JA, Watt DE: *Radiation Protection Dosimetry: A Radical Reappraisal*. Madison, WI, 1999, Medical Physics Publishing Corp.
4. Kondo S: *Health Effects of Low-Level Radiation*. Madison, WI, 1993, Medical Physics Publishing Corp.
5. Source and Effects of Ionizing Radiation (United Nations Scientific Committee on the Effects of Atomic Radiation, 2008 Report to the General Assembly), New York, 2010, United Nations. Available at http://www.unscear.org/unscear/en/publications.html [accessed October 14, 2011].
6. Sources and Magnitude of Occupational and Public Exposures from Nuclear Medicine Procedures (NCRP Report No. 124). Bethesda, MD, NCRP, 1996.
7. Anger RT: Radiation protection in nuclear medicine. In *The Physics of Nuclear Medicine*, Chicago, 1977, American Association of Physicists in Medicine.

BIBLIOGRAPHY

A detailed discussion of health physics and radiation protection techniques can be found in the following textbooks:

Shapiro J: *Radiation Protection: A Guide for Scientists, Regulators, and Physicians*, 4th ed. Cambridge, MA, 2002, Harvard University Press.

Turner JE: *Atoms, Radiation, and Radiation Protection*, 2nd ed. New York, 1995, Wiley.

Regulatory documents can be found on the website of the Nuclear Regulatory Commission at http://www.nrc.gov [accessed October 14, 2011]. Relevant documents on this website include the following:

NRC Regulations:

10CFR20: http://www.nrc.gov/reading-rm/doc-collections/cfr/part020/index.html

10CFR35: http://www.nrc.gov/reading-rm/doc-collections/cfr/part035/index.html

NRC Regulatory Guides (available at http://www.nrc.gov/reading-rm/doc-collections/reg-guides/occupational-health/rg/):

8.10: Operating Philosophy for Maintaining Exposures as Low as Reasonably Achievable

8.18: Information Relevant to Ensuring that Occupational Radiation Exposures at Medical Institutions will be As Low As Reasonably Achievable

8.36: Radiation Dose to the Embryo/Fetus

8.39: Release of Patients Administered Radioactive Materials

The National Council on Radiation Protection and Measurement website is at http://www.ncrp.com [accessed October 14, 2011]. Important NCRP publications in addition to those listed in the references are the following:

Uncertainties in the Measurement and Dosimetry of External Radiation (NCRP Report No. 158), 2007.

Operational Radiation Safety Training (NCRP Report No. 134), 2000.

Radionuclide Exposure of the Embryo/Fetus (NCRP Report No. 128), 1998.

Operational Radiation Safety Program (NCRP Report No. 127), 1998.

Limitation of Exposure to Ionizing Radiation (NCRP Report No. 116), 1993.

Implementation of the Principle of As Low as Reasonably Achievable (ALARA) for Medical and Dental Personnel (NCRP Report No. 107), 1990.

Radiation Protection for Medical and Allied Health Personnel (NCRP Report No. 105), 1989.

Protection in Nuclear Medicine and Ultrasound Diagnostic Procedures in Children (NCRP Report No. 73), 1983.

Management of Persons Accidentally Contaminated with Radionuclides (NCRP Report No. 65), 1980.

Safe Handling of Radionuclides (NCRP Report No. 30), 1964.

The International Atomic Energy Agency website is at www.iaea.org [accessed October 14, 2011] and has several publications relevant to nuclear medicine that can be downloaded from the website: For example:

Nuclear Medicine Resources Manual, 2006.

Cyclotron Produced Radionuclides: Guidelines for Setting Up a Facility, Technical Reports Series No. 471, 2009.

Quality Assurance for Radioactivity Measurement in Nuclear Medicine, Technical Reports Series No. 454, 2006.

Applying Radiation Safety Standards in Nuclear Medicine, Safety Reports Series No. 40, 2005.

International bodies providing information or recommendations regarding radiation dose:

The International Commission on Radiological Protection website is at http://www.icrp.org [accessed October 14, 2011). An important ICRP publication with recommendations regarding dose limits is the following:

Recommendations of the International Commission on Radiological Protection (ICRP Publication No. 103), Annals of the ICRP, Vol. 37, 2-4, 2007.

The United Nations Scientific Committee on the Effects of Atomic Radiation website is at www.unscear.org/ [accessed October 14, 2011). Two comprehensive publications of interest are:

UNSCEAR 2008 Report: "Sources and effects of ionizing radiation."

UNSCEAR 2006 Report: "Effects of ionizing radiation."