CHAPTER 13

Health Physics I

Health Physics deals with the subject of radiation protection to individuals employed in a radiation environment and the general public at large. In the medical environment, radiation used either in the diagnosis or treatment of diseases comes primarily from x-ray machines and radioactive sources. The radiation employed is of the ionizing type that can potentially produce undesirable health effects. Health physics teaches us how to manage and minimize unnecessary exposure to patients, staff personnel, and the general public. Staff personnel who are directly involved with ionizing radiation are technologists, therapists, medical dosimetrists, medical physicists, radiologists, nuclear medicine physicians, and radiation oncologists. The staff personnel trained in the operation of radiation producing machines and the safe handling of radioactive materials are referred to as **occupational workers**. Those personnel working around radiation sources such as housekeeping, maintenance, and security are referred to as allied medical workers or **ancillary medical workers**. Besides these identifiable personnel, other personnel who care for radioactive patients such as patient transporting staff, nurses, and operating staff may be exposed to radiation. This chapter will deal with radiation safety concerns related to the medical use of radioactive materials and radiation producing machines.

13.1 Equivalent Dose

The relative biological effectiveness (RBE) or destructiveness of radiation per unit absorbed dose varies with various types of radiation. For example, alpha rays are more destructive to living tissues than the same amount of absorbed dose from electromagnetic radiation. To take into account this RBE difference, an equivalent dose ($H_{T,R}$) quantity is defined for a radiation exposure from a radiation type, R, as the product of the radiation weighting factor, w_R times the average absorbed dose in a tissue or organ (T). Mathematically, the **equivalent dose** can be written as

$$H_{T,R} = w_R \times D_{T,R} \qquad (13.1)$$

where $D_{T,R}$ represents the average absorbed dose in the tissue or organ (T), w_R represents the radiation weighting factor associated with radiation type, R. This formalism was introduced in ICRP Publication 60 and later adopted into NCRP Report No 116. The w_R value is a dimensionless quantity that allows adjustment for the radiobiological destructiveness of the radiation. A higher w_R value implies that the radiation is considered more destructive compared to a lower w_R value. The w_R values for the various types of radiation are listed in Table 13.1.[1] In a radiation environment where there are various types of radiation with different energies such as nuclear reactors or particle accelerators, the equivalent dose is determined as the sum of the contributions from all types of radiation. This sum of all types of radiation can be written as

$$H_T = \sum (w_R \times D_{T,R}) \qquad (13.2)$$

where the Greek letter symbol Σ, pronounced as "sigma", used to indicate the summation of the products.

Table 13.1 Radiation weighting factors

Radiation Types	Weighting factor (w_R)
x-rays, gamma rays, electrons, positron, and muons	1
Neutrons < 10 keV	5
10 keV to 100 keV	10
> 100 keV to 2 MeV	20
> 2 MeV to 20 MeV	10
> 20 MeV	5
Protons, other than recoil protons with energy > 2 MeV	2[a]
alpha particles, fission fragments non-relativistic heavy nuclei	20

[a]ICRP Pub No. 60 lists a value of 5.

EXAMPLE 13.1 If a radiation worker receives an average whole body dose of 100 mrad from x-rays and 5 mrad from 2 MeV neutrons, what is the dose equivalent to this person?
SOLUTION:

$$H_T = \Sigma(w_R \times D_{T,R}) = (100 \times 1) + (5 \times 20) = 200 \text{ mrem}$$

As defined, the equivalent dose (H_T) quantity involves both radiobiological response and physical quantities. The unit of equivalent dose is either the **rem** (**r**ad **e**quivalent **m**an) or sievert (Sv) depending on the unit of absorbed dose used in equation (13.1). If the absorbed dose unit is expressed in rad, the equivalent dose unit is the rem. The sievert (Sv) is the SI unit for the equivalent dose when the absorbed dose is expressed in gray (Gy). The General Conference on Weights and Measures approved the use of sievert as the equivalent dose unit in 1979. One Sv is equal to 100 rems or conversely **1 rem is equal to 10 mSv**. In the special case of exposure to x-rays, gamma rays, and electrons where the weighting factor is 1, the dose equivalent and absorbed dose are considered the same, that is,

[1] NCRP Report No. 116. *Limitation of Exposure to Ionizing Radiation*. Bethesda, (MD): National Council on Radiation Protection and Measurements; 1993.

$$1\ \text{rem} = 1\ \text{rad}$$
$$1\ \text{Sv} = 1\ \text{Gy} = 100\ \text{rad}$$

Keeping this in mind, the radiological units of rem, rad, and R are sometimes used interchangeably for radiation protection purposes in medical facilities in spite of the differences in their meaning.

EXERCISE 13.1 Express the solution (200 mrem) in Example 13.1 in Sv.

13.2 Environmental Radiation

Radiation is everywhere; it is being emitted from the earth and coming from space, as well as from artificially produced radionuclides and radiation generating machines. The radiation that is naturally present in the environment is referred to as natural or **background radiation**, and together with the radiation from man-made sources, constitutes the **environmental radiation**. There are three basic sources of background radiation. **Cosmic radiation** (or cosmic rays) comes directly from outer space and from its interaction with the nuclei in the Earth's atmosphere. The radiation that comes directly from outer space consists mainly of high-energy protons, alpha particles, atomic nuclei, high-energy electrons, and photons are referred to as **primary cosmic rays**. The next radiation source is the **secondary cosmic rays** produced through the interaction of the primary cosmic rays and the Earth's atmospheric nuclei. It consists mainly of mesons (mostly π-mesons but also include μ-, τ-, and κ-mesons), hydrogen-3 (^{3}H), electrons, and gamma rays. The π-mesons, often simply called pions, have a short lifetime of about 10^{-6} seconds and are observed in three charge states: $+1$, 0, or -1. Compared to the mass of the electron (m_e), both π^+ and π^- have a rest mass of about 273 m_e while the π^0 has a rest mass of about 264 m_e. Cosmic radiation detected in the Earth's atmosphere is mostly of the secondary type, which is highly energetic and can penetrate several meters of lead (Pb). **Terrestrial radiation** comes from radioactive materials that occur naturally in the Earth's crust such as radon-222 gas, radium-226, carbon-14, and potassium-40. Release of radiation can come from earthquakes and volcanic eruptions as well. **Internal radiation**, which arises from radioactive materials such as potassium-40 and carbon-14 are present in living organisms. These radioactive materials are either present at birth or ingested from the food supply or inhaled from the air as we breathe. In general, the quantities of these radionuclides are very small. Only radioactive potassium-40 present in our diet shows an appreciable contribution to human exposures. The estimated amount of radiation from the different sources is given in Table 13.2.[2] **Natural radiation exposure is about**

[2] NCRP Report No. 93. *Ionizing Radiation Exposure of the Population of the United States*. Bethesda, (MD): National Council on Radiation Protection and Measurements; 1987.

3 mSv per year of which, 2 mSv per year is from radon gas. Exposure from man-made radiation is about 0.6 mSv per year and mostly from medical procedures.

The amount of radiation exposure composition from natural sources varies considerably depending on the altitude, latitude, and the amount of radioactivity in the Earth's crust. Exposure from cosmic radiation increases with altitude (higher elevation) and latitude (the polar region). Hence, travel by airplane exposes the crew and passengers to higher level of cosmic radiation. Natural background radiation originating from the Earth's crust, for example, at the Colorado Plateau area (near Denver) where there are extensive deposits of uranium has an average background radiation of 0.63 mSv per year.[3] This exposure is higher than the background radiation of the middle-America region (0.46 mSv) or the Atlantic and Gulf Coast Plain (0.16 mSv) in the United States. Still higher radiation exposure, about 10 times more than the average in the United States occurs in Kerala, India due to the high thorium content in the soil. In general, the average cumulated exposure from background radiation is about 70 mSv to a 20 year-old person, 180 mSv by the age of 50, and 250 mSv by the age of 70.

Table 13.2 Estimated environmental radiation sources

Radiation Sources	Exposure (mSv/yr)
Natural	
Cosmic rays	0.26
Terrestrial radionuclides	0.28
Internal radionuclides	~0.40
Radon	2.00
Man-made	
Medical	0.54
Consumer products	0.05 – 0.13
Fallout	0 .0005
Nuclear power plant	< .001

The largest contributor of natural radiation to the population in the United States is radon, an inert but radioactive gas. In addition to being inert, it is also invisible and has neither smell nor taste. Radon gas is produced through the natural decay of uranium that is present in trace quantities in the Earth's crust. Because of the structural design, some buildings have elevated radon gas that is emitted from uranium-238 present in the soil at many locations in the United States. Radon gas emits alpha particles, which do not penetrate more than a few millimeters of tissues. Its main effect relates to the damage of the mucosa region in the lungs. In fact, the inhalation of radon gas is the second leading cause of lung cancer.

The largest source of environmental radiation is from man-made sources used in medical procedures. Sources of radiation used in medicine include x-ray machines, radiation treatment machines, and radioactive materials used in the diagnosis and treatment of diseases. Some radioactive materials are artificially produced for specific medical procedures. Once the radioactive materials are produced, they will continuously emit radiation and decrease over time according to their half-life. **The radiation from radioactive materials cannot be turned off**. These radioactive materials should be stored in shielded areas to minimize radiation exposure to the public. For those patients receiving internally administered radioactive therapy, the patients are

[3] NCRP Report No. 93. *Ionizing Radiation Exposure of the Population of the United States*. Bethesda, (MD): National Council on Radiation Protection and Measurements; 1987.

considered temporarily radioactive until the radioactive material decays to an acceptable level or is eliminated naturally (most often through urination).

Other sources of man-made radiation are from the testing and the use of nuclear weapons, nuclear reactors used to generate electricity, consumer items, and the industrial use of radiation such as the irradiation of food for preservation. At the height of atmospheric nuclear weapon testing in the early 1960s, nuclear fallout contributed about 0.05 mSv per year to radiation exposure. Today, the fallout exposure is less than 0.01 mSv. The radiation exposure contributions from nuclear power stations and other industrial applications are insignificant. Consumer and electronic products such as watch dials, television, computer terminals, smoke detectors, and airport surveillance systems including air baggage inspection machines, also contribute to environmental radiation. However, the amount of radiation exposures from these sources is also small. While natural background radiation is unavoidable, radiation from man-made sources can be controlled to some extent.

13.3 Radiation Advisory Boards

For many years after their discovery, x-rays and radioactive materials were used with little regard for their potentially harmful biological effects. Many early pioneers involved in the application of radiation in medicine died from the late effects of radiation exposure. Madame Curie and her daughter are both suspected to have died of leukemia as a result of radiation experiments. Early physicians and health care workers also used x-ray producing machines that lacked shielding. Furthermore, patients had to endure lengthy radiation exposures from low energy x-rays produced by x-ray machines to obtain acceptable radiographs. These long exposures occasionally lead to severe skin burns, loss of hair, nausea, and anemia, and after some time, leukemia and other cancers.

Because of the risks associated with radiation injury, advisory boards such as the **International Commission of Radiation Protection (ICRP)** and the **National Council on Radiation Protection and Measurements (NCRP)** were formed to develop standards of radiation protection. The Second International Congress of Radiology and the International Commission on Radiation Units and Measurements (ICRU) together, established the ICRP in 1928. The ICRP often takes the lead in formulating concepts in radiation protection and making recommendations on dose limits. These councils are composed of scientific experts in this field with their interests in the safe use of radiation. Because these councils are advisory bodies, they merely issue recommendations, and do not make regulations.

The philosophy of these groups is to establish upper limits of radiation exposure that would theoretically minimize the radiation hazards to individuals and the population as a whole but not encumber the beneficial use of radiation. These groups take into consideration social and economic aspects of protection (e.g., shielding materials, monitoring devices, personnel restrictions, and remote control techniques). Risk estimation for ionizing radiation has been published by the United Nations Scientific Committee on the Effects of Atomic Radiation (UNSCEAR), National Academy of Sciences, National Research Council, Committee on the Biological Effects of Ionizing Radiation (BEIR), and National Institute of Health (NIH). One of the most recent and widely quoted risk estimates is from Report No. 5 of the National Academy of Sciences Committee on Biological Effects of Ionizing Radiation (BEIR V), 1990. The BEIR V report used the linear quadratic model for leukemia estimates and the linear dose-effect model for all other cancers.[4] The effect of age at the time of irradiation and the elapsed time since irradiation were also considered. There was an assumed latency period of 2 years after irradiation for leukemia, and after 10 years for all other cancers. The relative-risk model of cancer incidence as a function of age based on the recent trends in stochastic effects (e.g., the probability of such effects increases with increasing radiation exposure but not the severity of the effects) among the atomic bomb survivors was also included in the estimation.

In practice, the underlying radiation protection principle is to keep the radiation exposure to individuals "**as low as reasonably achievable**". This principle of practice is often referred to as the **ALARA** principle. It neither establishes a lower limit of radiation exposure nor provides a means of lowering radiation exposure.

In 1964, congress chartered the NCRP to interpret the recommendations of the ICRP for the United States**. The NCRP operates as an advisory body without legal status**. However, its recommendations are often adopted as regulatory guidelines in the United States by federal and local government agencies including the United States Food and Drug Administration (FDA), United States Nuclear Regulatory Commission (NRC), and United States Environmental Protection Agency (EPA). As such, the reports from the NCRP may serve as a preview of current and future changes in regulatory guidelines and in particular, the dose limits for radiation protection. Handbooks on the recommendations of the NCRP are published periodically. When the NCRP was established in 1931, it was named as Advisory Committee on X-ray and Radium Protection.

The philosophy in establishing the dose limits has changed over the years. In the 1930's, the dose limit for occupational workers was based on the concept of **tolerance dose**. The tolerance dose is the dose that a worker could tolerate when exposed continuously, without suffering from deleterious acute symptoms or early effects of radiation such as skin erythema.[5] In the 1950s, the emphasis had shifted to chronic or late effects such as leukemia. The **maximum permissible dose** (MPD) was established so that the exposure to such a dose would result in a very small fatal cancer risk and would be hidden within the normal biological variations. With the availability of genetic risk

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[4] Committee on the Biological Effects of Ionizing Radiation (BEIR V). *Health Effects of Exposure to Low Levels of Ionizing Radiation*. National Academy of Sciences / National Research Council. Washington (DC): National Academy Press; 1990.

[5] NCRP Report No. 39. *Basic Radiation Protection Criteria*. Bethesda, (MD): National Council on Radiation Protection and Measurements; 1971. pp. 61-63.

results, the occupational dose limits were reduced substantially and public dose limits were introduced. The genetic risks were subsequently found to be smaller, while the cancer risks were larger than expected. Today, the dose limits are recommended based on the comparison of the **probability of radiation-induced cancer mortality to the annual accidental mortality in safe industries**. The mean fatal accidental rate in safe industries was stated to be 0.90×10^{-4} per year according to the 1991 data published by the National Safety Council. The NCRP recommended that an annual average occupational dose limit of 10 mSv per year would yield a cumulative risk between 1.0×10^{-4} and 1.0×10^{-3}. Such a dose limit would yield a lifetime risk for each year's exposure from 2×10^{-5} to 2×10^{-4} to an average radiation worker.[6] With the changes in philosophy, the acceptable radiation exposure dose limits have been modified accordingly and are called **effective dose limits**.

In developing the upper dose limits of ionizing radiation exposure to personnel, the ICRP and NCRP divided the population into occupationally exposed individuals and members of the general public. Furthermore, the upper dose limits exclude environmental radiation and medical exposures. Radiation exposures to patients undergoing medical procedures have been addressed by the FDA, NRC, and EPA. A summary of current recommendations on the exposure dose limits from NCRP Report No. 116 is given in Table 13.3.[7] These values constitute the sum of the external and internal exposures, but exclude exposure from natural radiation. Exposures from medical procedures are not included in these dose values.

As shown in Table 13.3, **effective dose limits (EDL)** are used in the current recommendation for personnel exposure instead of the **maximum permissible dose (MPD)** as cited in NCRP Report No.

Table 13.3 Dose limit recommendations

A. Occupational exposures	
1. Effective dose limits	
a) Annual	50 mSv
b) Cumulative	10 mSv x age
2. Equivalent dose annual limits for tissues and organs	
a) Lens of eye	150 mSv
b) Skin, hand and feet	500 mSv
B. Public exposures (annual)	
1. Effective dose limit, continuous or frequent exposure	1 mSv
2. Effective dose limit, infrequent exposure	5 mSv
3. Equivalent dose limits for tissues and organs	
a) Lens of eye	15 mSv
b) Skin, hands, and feet	50 mSv
C. Education and training exposures (annual)	
1. Effective dose limit	1 mSv
2. Equivalent dose limits for tissues and organs	
a) Lens of eye	15 mSv
b) Skin, hand and feet	50 mSv
D. Embryo-fetus exposures (monthly)	
1. Equivalent dose limit	0.5 mSv
E. Negligible individual dose (annual)	0.01 mSv

[6] NCRP Report No. 116. *Limitation of Exposure to Ionizing Radiation*. Bethesda, (MD): National Council on Radiation Protection and Measurements; 1993. pp. 14-15.
[7] Ibid.

39.[8] Both terms represent the upper dose limits, but their derivations were based on two different philosophies as described above.

In the case where the exposure is not uniform, the concept of equivalent dose to an organ or tissue is used. The effective dose (E) is the product of the equivalent dose and the tissue or organ weighting factor (w_T). The tissue or organ weighting factor accounts for the difference in sensitivity of each tissue or organ to radiation. The organ or tissue weighting factors from NCRP Report No. 116 adopted from ICRP Publication 60 are given in Table 13.4. The effective dose (E), which is the sum of dose to all tissues and organs, can be written as

$$E = \sum (w_T \times H_T) \qquad (13.3)$$

where H_T is the equivalent dose given in equation (13.2). Note that there is a difference between w_T and w_R, radiation weighting factor. The **w_T is only dependent on the organ or tissue type** and independent of the radiation type or energy. On the other hand**, w_R is only dependent on the radiation type and energy** and independent of the organ or tissue types.

Table 13.4 Organ or tissue weighting factor (w_T)

Tissues/Organs	Weighting Factors
Gonads	0.20
Red bone marrow	0.12
Colon	0.12
Lung	0.12
Stomach	0.12
Bladder	0.05
Breast	0.05
Liver	0.05
Esophagus	0.05
Thyroid	0.05
Skin	0.01
Bone surface	0.01
Remainder	0.05
Total	1.00

The occupational annual EDL for a radiation worker is set at 50 mSv while for the general public, it is set not to exceed one-tenth of this value for infrequent exposure. Infrequent exposure means exposure to persons who are not routinely present near the radiation environment, such as visitors to a patient receiving radioactive implants. On the other hand, the EDL for those who are routinely subjected to continuous or frequent exposure, such as nursing staff is set at 1 mSv. It should be reiterated that the NCRP reports are merely guidelines, and that current regulations in each state vary. For example, the state of Nebraska sets its own dose limit to the general public as 1 mSv regardless of whether the exposure is classified as frequent or infrequent exposure. The fractional representation is also applied for equivalent dose limits for organs and tissues. For example, the dose limit to the lens of the eye for the general public is one-tenth the dose limit for occupational workers. Higher dose limits are set for some organs and areas of the body considered less sensitive to radiation. For example, the equivalent dose limit for the lens of the eye is 15 mSv, while for the skin, hands and feet, it is set at 50 mSv for public exposures.

The cumulative lifetime effective dose limit for a radiation worker is set not to exceed 10 times the age (in years) of the individual. This lifetime limit is not implemented in current state regulations. Sometimes, the cumulative

[8] NCRP Report No. 39. *Basic Radiation Protection Criteria*. Bethesda, (MD): National Council on Radiation Protection and Measurements; 1971. p.106.

effective dose limit and the annual effective dose limit may not be satisfied simultaneously.

EXAMPLE 13.2 A 23-year-old radiation worker has been exposed to the effective dose limits annually since the age of 18. What is the worker's total cumulated dose in mSv? What is the cumulative dose limit in mSv?
SOLUTION:

a) The worker's total cumulated dose is

$$X_T = 50\frac{\text{mSv}}{\text{yr}} \times (N-18)\ \text{yr}$$

$$= 50\ (23-18)\ \text{mSv} = 50\ (5)\ \text{mSv} = 250\ \text{mSv}$$

b) The cumulative dose limit is

$$X_C = 10\ \text{mSv} \times \text{Age (in years)} = 10\ \text{mSv} \times 23 = 230\ \text{mSv}$$

The above example shows that the total dose exceeds the recommended cumulative effective dose limit. The conservative recommendation is intended to ensure that career exposure is kept as reasonably low as achievable. In 1987, a new formula was introduced in NCRP Report No. 91 that supercedes the previously used formula of 5(N-18) rem, where N is the age in years in NCRP Report No. 39.

The radiation exposure to students under the age of 18 who are undergoing their educational or training activities should not exceed 1 mSv per year. The effective dose limit for the students is identical to that for the public. Although exposure to minors under current regulations is one-tenth of the occupational values (5 mSv), child labor laws may restrict minors from working with radioactive materials and ionizing radiation.[9]

For a pregnant radiation worker, the embryo-fetus is not considered as a radiation worker, and hence is subjected to a more stringent requirement than the occupational exposure limit. According to NCRP Report No. 91, the **total exposure limit to the embryo-fetus is 5 mSv for the entire pregnancy, and the exposure per month should not exceed 0.5 mSv**. The total exposure limit is not mentioned in the NCRP Report No. 116. In most instances, when the pregnancy is made known, the duties of the pregnant worker are modified so that the exposure limit cannot be reached. Such measures comply with the ALARA principle.

The Federal Emergency Management Agency (FEMA) has set guidelines of exposure dose limits for emergency occupational exposures. This agency was formed in 1979 under the Carter Administration and now has become part of the new Department of Homeland Security in March 2003. The role of this agency is to prepare for, respond to, and recover from disasters. Acute exposure significantly over the annual effective dose limit is generally not justifiable except to save lives. In the case of lifesaving and protection of large populations, the dose limit was set at 0.25 Sv by FEMA. For radiation exposure exceeding 0.25 Sv in a short time, the workers need to understand

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[9] NRC Regulations Part §20.1207. *Standard for Protection Against Radiation – Occupational dose limits for minors*.

the risk of acute effects as well as the substantial increase in lifetime risk of cancer. With this in mind, volunteers may be used. Older workers with low cumulative effective dose limits should be chosen among the volunteers, whenever possible. In emergency procedures not involving lives (property only), the radiation exposure should be controlled to less than 0.10 Sv.

In 1987, the NCRP Report No. 91 introduced the concept of **Negligible Individual Risk Level (NIRL)** as the level of average annual excess risk of fatal health effects attributable to radiation below which efforts to reduce radiation exposure to the individual is unwarranted. In other words, the risk associated with radiation exposure below the NIRL is dismissed from consideration. The NIRL should not be interpreted as having no risk, an acceptable risk level, or a lower limit. To reiterate, it is merely a value below which the risk should not be considered for further assessment. The establishment of a value for NIRL merely presents a low-level comparable risk that is being regarded as safe in a radiation workplace. This risk is being set at 1.0×10^{-7} per year, corresponding to an annual effective dose equivalent of approximately 0.01 mSv.

13.4 States and Federal Regulatory Agencies

In the United States, **the United States Nuclear Regulatory Commission (NRC)** and the individual state regulatory agencies establish regulations, which are incorporated into state and federal laws. The NRC formally known as the **Atomic Energy Commission (AEC)** was established as an aftermath to the World War II atomic bomb project to regulate the use of **all reactor-produced (or byproduct) materials**. Examples of reactor byproduct materials include cobalt-60, iodine-125, technetium-99, cesium-137, and iridium-192. Through an agreement with the NRC, some states referred to as **agreement states** are allowed to govern these byproduct materials regulated by the NRC. The agreement states are shown in Figure 13.1 as shaded states.[10] The use of radioactive materials other than reactor byproduct materials, and radiation producing equipment such as linear accelerators,

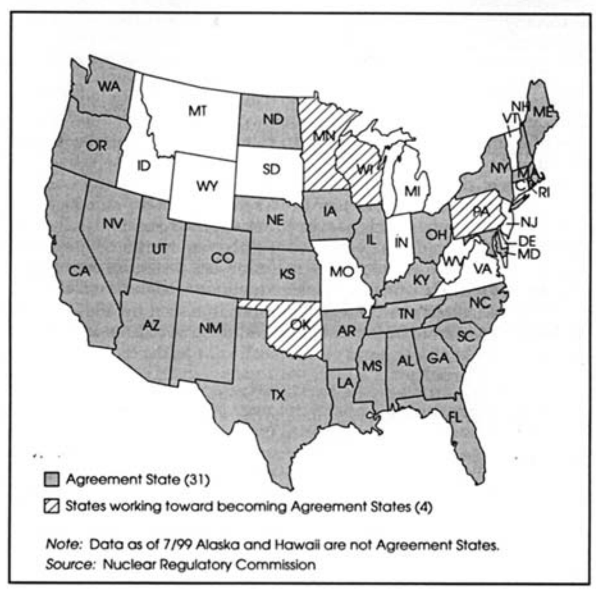
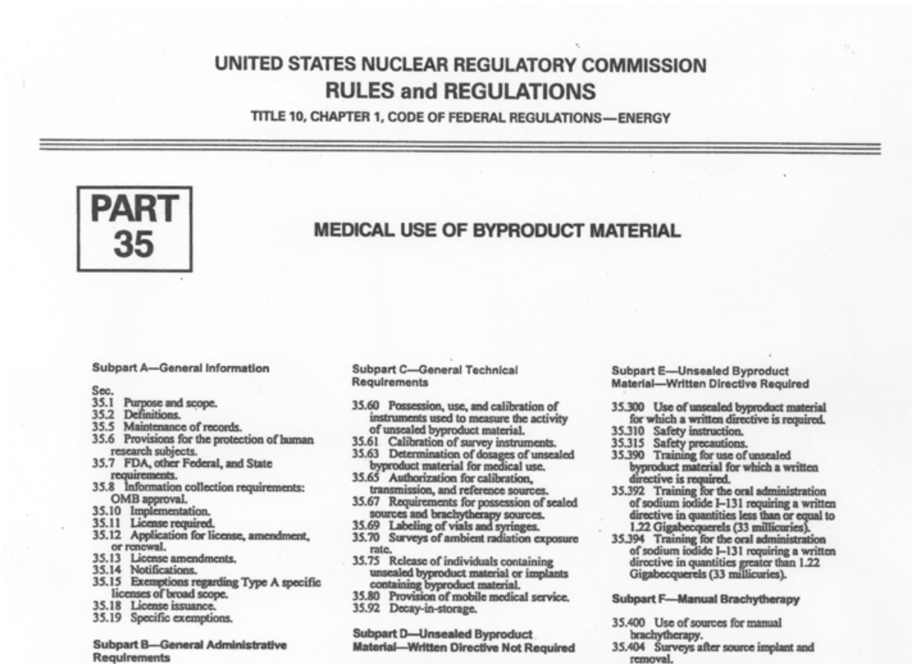


Figure 13.1 NRC agreement states

[10] NUREG/BR-0256. *The US Nuclear Regulatory Commission and How It Works*. Office of Nuclear Material Safety And Safeguards. US Nuclear Regulatory Commission, Washington, DC 20555, 2000; p. 29.

particle accelerators, and x-ray machines are regulated by the individual state regulatory agencies. Examples of these radioactive materials that are not regulated by the NRC are fluorine-18, thallium-201, gallium-67, radon-222 and radium-226. As mentioned in Section 8.6, radon and radium are naturally occurring radionuclides. The NRC regulations are contained in the Title 10, Code of Federal Regulations (CFR), parts 19, 20, and 35. A front portion of CFR part 35 of the NRC is shown in Figure 13.2.



UNITED STATES NUCLEAR REGULATORY COMMISSION
RULES and REGULATIONS
TITLE 10, CHAPTER 1, CODE OF FEDERAL REGULATIONS—ENERGY

PART 35

MEDICAL USE OF BYPRODUCT MATERIAL

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Subpart F—Manual Brachytherapy

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35.404 Surveys after source implant and removal.

Figure 13.2 Sample page of the NRC regulations

These regulations are based on the recommendations of the advisory bodies such as the NCRP, ICRP, and AAPM (American Association of Physicists in Medicine). However, some of the regulations have been enacted in direct response to actual radiation incidents. All new regulations must undergo public and congressional hearings as part of the federal government processes before enactment. The NRC had also enacted a quality management program to minimize radiation incidents, in particular those due to human errors. Again, this enactment was based on the assessment of past incidents reported to the NRC.[11]

The requirements of notifying either the state or NRC of any adverse radiation events depend on the severity of the incident. **Notification must be immediate for serious events** such as the loss of a radiation source in quantities deemed hazardous to the public. The notification is typically followed by a written report within 30 days. **Notification must also be immediate if an individual receives exposures exceeding the dose limits stated in the regulations**. Less severe events require notification within 24 hours. Notification must also be made in the event of misadministration,

[11] Redmond, P. *Nuclear Regulatory Commission Report.* Med. Dosm. 13: 87-93; 1988.

referred to as a medical event.[12] Currently there are six categories of medical events. This class of medical events will be discussed in the Quality Management Program section of Chapter 14.[13]

As stated above, the **NRC maintains a reporting system in the event an incident occurs**. The NRC usually investigates the event and formulates a report. The report is presented to the "offending" institutional authorities for further action. Meanwhile, the institutional personnel charged with overseeing the use of radioisotopes and radiation from byproduct materials, typically the radiation safety officer and/or medical physicist, perform their own investigations to determine if there is any procedural weakness in the program, which led to the incident. Corrective action is proposed to ensure that such an incident will not recur. The final step is either an admission of the weakness of the program followed by corrective action, or a consultation with the NRC leading to taking the appropriate steps to ensure the incident will not occur again. Depending on the type and severity of the infraction, a fine may be imposed by the NRC. Since the NRC is obliged to inform the public, the fear and lack of understanding of radiation often leads the public to adopt pessimistic viewpoints of the use of radiation. One of the most eventful accidents occurred in November 1992 at Indiana Regional Cancer Center in Pennsylvania involving high dose rate (HDR) brachytherapy. A high intensity iridium-192 source broke off during a rectal radiation treatment. The broken source was not detected until the death of the patient in the nursing home. The NRC publication report NUREG-1480 documented this incident. Any radiation incident not involving the use of byproduct materials has to be reported to the local or state agencies. Procedures identical to those used by the NRC are generally adopted by the local or state agencies.

13.5 Radiation Detection Instruments

All electromagnetic (EM) radiation except for visible light cannot be readily detected by our senses. Special instruments are used to detect the presence of invisible EM radiation based on their interaction effects with matter. The many interaction effects of ionizing radiation are identified in Chapter 1. However, photographic, thermoluminescence, and ionizing of gas effects are employed in the design of radiation detection instruments used for radiation protection purposes.

The simplest form of radiation detection device is x-ray film. The film consists of a transparent base and emulsion. The transparent base provides the structural rigidity while the emulsion interacts with the radiation. Today, the base is made of polyester instead of cellulose triacetate or cellulose nitrate, which have been used in the past. Cellulose nitrate, a flammable product, was discontinued in favor of cellulose triacetate. The emulsion is composed

[12] NRC Regulations Part §35.3045. *Medical uses of byproduct materials – Report and notification of a medical event.*

[13] NRC Regulations Part §35.32. *Medical uses of byproduct materials – Quality management program.*

of a homogeneous mixture of gelatin and very small silver halide crystals (silver bromide and silver iodide). When film is exposed to radiation, the affected crystals are converted into metallic silver. During development, the unaffected portion of the emulsion is washed off. The remaining metallic silver causes the darkening of the film. The degree of darkening is proportional to the amount of radiation energy absorbed.

The degree of film darkening, referred to as **optical density**, is measured using a densitometer. The densitometer consists of a light source directed through a small aperture towards a light detector. As designed, the transmitted light intensity is measured when the film is placed in the pathway of light. The optical density is defined as the logarithm of the ratio of the transmitted intensity without and with the film in place as

$$OD = \log_{10}\left(\frac{I_o}{I_t}\right) \tag{13.4}$$

where I_o and I_t is the transmitted intensity without and with film in place, respectively.

EXAMPLE 13.3 After a film has been exposed to 100 cGy and processed, the optical density was found to be 1.5. What is the transmitted intensity?
SOLUTION:

$$\left(\frac{I_t}{I_o}\right) = 10^{-OD} = 10^{-1.5} = 0.0316$$

The transmitted intensity is 3% of the original intensity.

Film fog density, due in part to background exposure in storage and development contaminants, is subtracted out before calculating the optical density. A plot of the net optical density versus the radiation exposure or dose is called the sensitometric curve or **H&D curve**, named after Hurter & Driffield for three types of films is shown in Figure 13.3.[14] If the optical density of a film exposed to radiation is known, the amount of

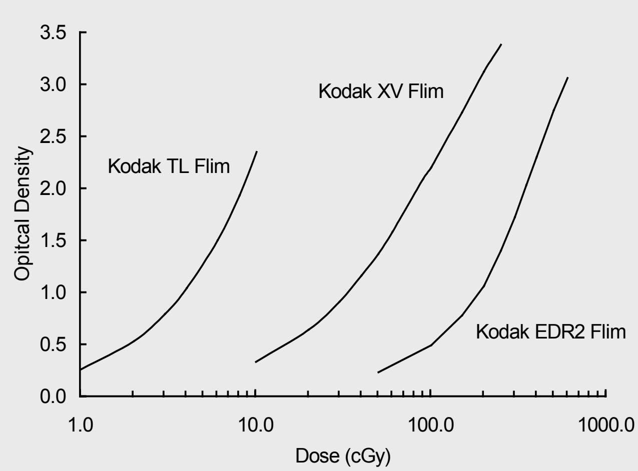

Figure 13.3 Dose versus optical density

[14] Saw, C.B.; Li, SiCong; Ayyangar, K.M.; Yoe-Sein, M.; Pillai, S.; Enke, C.A. *Dose linearity and uniformity of a linear accelerator designed for implementation of multileaf collimation system-based intensity modulated radiation therapy*. Med. Phys. 30: 2253-2256; 2003.

radiation exposure can be determined from this curve.

The next type of radiation detection device is the **thermoluminescent dosimeter**, also called **TLD**. Thermoluminescent materials operate on the principle of converting ionizing radiation energy into light energy as illustrated in Figure 13.4. Theoretically, the thermoluminescent material is assumed to be an imperfect crystal. When exposed to ionizing radiation with sufficient energy, electron-hole pairs are created. The result is the excitation of crystal with electrons moving into the excitation or conduction band from the valence band of the crystal. During de-excitation, some of the electrons are trapped in higher energy states, which cannot decay to the valence band unless they are excited again into the conduction band. Heat is used to excite the trapped electrons into the conduction band. These electrons emit energy in the form of light when making transitions to the valence band. The number of electrons trapped is proportional to the total energy received from the ionizing radiation. To retrieve this proportionality information, the TLD chips are initially placed in a thermoluminescent dosimeter reader and subjected to heat. The light emitted from the chips is integrated and processed into electrical signal. The amount of light emitted is proportional to the amplitude of the signal and also to the amount of radiation exposure. The most commonly used TLD materials in radiation protection are **lithium fluoride** (LiF) and **manganese activated calcium fluoride** (CaF_2:Mn) chips. The radiation exposure that can be measured using TLD dosimeters ranges from 1 mR to 100,000 R (0.01 mSv to 1000 Sv).

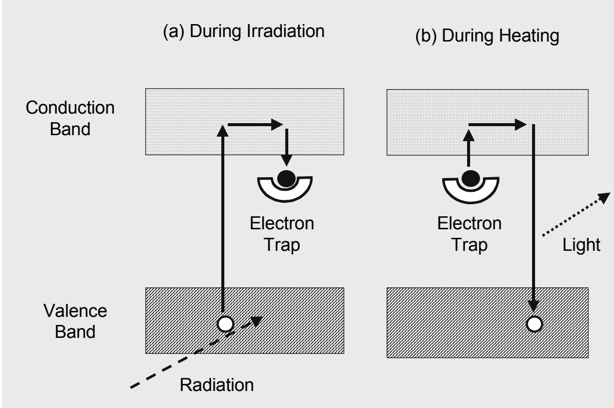

Figure 13.4 Schematic of TLD excitation

Recently, a new type of measuring device based on **optically stimulated luminescence** (OSL) technology for radiation protection dosimetry has been introduced. The monitoring device, which is made of carbon-doped aluminum oxide (Al_2O_3) strip, is exposed through an open window, tin filter, and copper filter. The principle of operation assumes that the material has light-sensitive traps referred to as shallow, dosimetric, and deep traps. The shallow trap is closer to the conduction band while the dosimetric and deep traps are closer to the valence band. During irradiation, the dosimetric and deep traps will be populated with electrons. To perform the readout, the material is stimulated with a pulsed optical laser causing the trapped electrons in the dosimetric trap and deep trap to jump into the conduction band. The luminescence from the material during de-excitation of the electrons to the valence band is proportional to the amount of radiation. This type of dosimeter is commercially available under the trademark "LUXEL".[15] This

[15] Trademark of Landauer, Inc.

dosimeter has a wide dynamic exposure range, high sensitivity, and provides permanent record. The response range of the photon beam is from 1 mrem to 1000 rem; of beta particles from 10 mrem to 1000 rem; and of neutrons from 20 mrem to 20 rems.

The other types of radiation detection instruments are gas-filled detectors. The different types of gas-filled detectors are ionization chambers, proportional counters, and Geiger-Muller detectors. The basic operation of these detectors is illustrated in Figure 13.5. The detector consists of a cylindrical metallic tube filled with a certain type of gas. As radiation passes through the tube, it ionizes atoms of the gas along its path creating ion pairs. If these ion pairs are not separated, they will subsequently recombine. To separate the ion pairs, a bias voltage is applied across the tube between the wall of the chamber and the central electrode. Positive ions are attracted to the negative cathode while negative ions are attracted to the positive anode by the negative bias voltage. **Saturation** refers to the condition where all the ions produced are collected before recombination occurs. The positive anode is the wire running along the center and extending from one end to the other end of the cylindrical chamber. The movement of the ion pairs creates an electrical signal or pulse, which can then be measured. The magnitude of the measured pulse is proportional to the radiation intensity that creates the ion pairs.

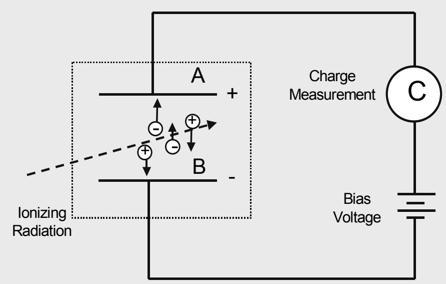

Figure 13.5 Basic operation of gas-filled detector

The difference between the three types of gas-filled detectors is the magnitude of the bias voltage applied across the tube. If the bias voltage is increased slowly from zero, the current response will increase in stages. In region I of Figure 13.6, there will be no observed current since ion pairs recombine at low bias voltage. This region is referred to as **region of recombination**. As the bias voltage is increased up to a certain level, all ion pairs released will move to the electrodes. The bias voltage at which this event occurs is typically from 100 to 300 volts for medical equipment. This operating region (region II) is called the **ionization region**. Ion chambers and radiation survey instruments such as the portable "cutie pie" operate in this region. As the bias voltage is further increased, the ion pairs will move faster towards the electrodes causing secondary ionization. The occurrence of secondary ionization increases with increasing bias voltage. The operation in this region

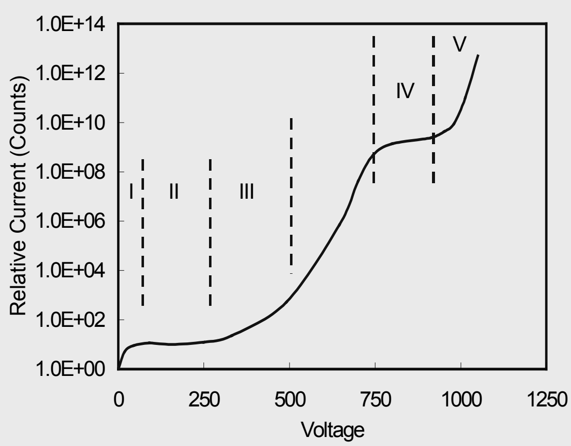

Figure 13.6 Operational region of gas-filled detector: II–Ionization region, IV–GM counter region

III is referred to as the "**proportional region**". Beyond the proportional region is region IV, which is the "**Geiger-Muller (GM) region**". In this region, the bias voltage is very high causing an avalanche of secondary ionization from a single ion pair. If the ionizing events occur too quickly, it is possible for the detector not to be able to detect the next event or pulse. To assist the detector to return to its original condition, a quenching agent (usually a halogen gas) is added to the gas in the chamber. The minimum time required in resolving two ionization events is called the **resolving time**. Because of the avalanche effect, a Geiger-Muller (GM) counter is very sensitive to radiation. As such, a Geiger Muller counter is excellent in detecting minute amounts of radiation. Beyond the Geiger-Muller region is the **region of continuous discharge** (region V). In this region, electrons from the gas discharge continuously and hence, this region is not used in a radiation detector.

In addition to the stated detection systems, there are also scintillation detectors. Whenever the scintillating material in the detector is exposed to radiation, it immediately fluorescence and emits light that is proportional to the amount of radiation absorbed. If the light emitted is directed towards a photomultiplier tube, the light signal is converted into current for measurement. Scintillation counters such as **sodium iodide well counters** are very sensitive and are used primarily to measure minute amounts of radioactivity in a wipe test. Low energy beta rays from radionuclides such as tritium (^3H) are detected using a **liquid scintillator** such as toluene where the radioactive source is mixed into the liquid.

Neutrons cannot be detected directly because they are indirectly ionizing. The detection method is based on the collision of neutrons with a gas that produces charged recoiled atomic nuclei. For example, the portable neutron rem counter, which is a proportional ionization chamber, is composed of a detector filled with boron fluoride (BF_3) gas. The neutrons interact with boron-10 to produce lithium recoil nuclei and alpha particles. Both of these particles ionize the gas in the chamber, providing the means of detecting neutrons. A large polyethylene cylinder is used around the detector to slow down fast neutrons for detection. Another type of neutron detection counter is the fission chamber. It uses the fission principle where one electrode is coated with uranium-235 mixed with a phosphor. When the neutrons interact with the mixtures, fission particles are emitted. The emitted particles strike the phosphors, causing them to scintillate and produce light, which is measured using a photomultiplier tube. A third method is photographic film. Recoil nuclei produced by the interaction of neutrons with low atomic number elements are detected by their tracks in photographic films.

Commonly used radiation detection systems for surveying, monitoring, and measuring of radiation exposure are shown in Figure 13.7. Item A in the figure is the neutron counter used for neutron measurements. The item labeled B is an ion chamber survey meter and item C is the GM counter. Technically, the GM counter cannot be used for radiation measurement because of the amplification of the detected signal. However, if it is properly calibrated, the GM counter can give acceptable readings. Items D, E, and F are a digital pocket dosimeter, ring badge and body badge, respectively for personnel monitoring. Lastly, item G is a scintillation counter where the detection system is sodium iodide crystals.

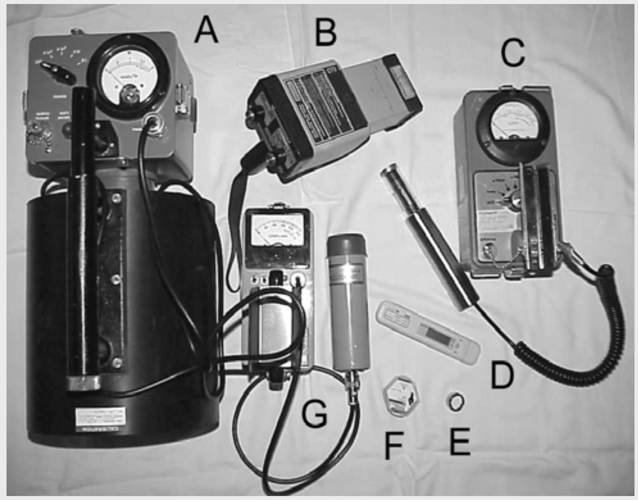

Figure 13.7 Commonly used radiation detection instruments

Films are commonly used for personnel monitoring and on a limited basis for monitoring an area. TLD is a useful dosimeter for monitoring places that are difficult to access. A scintillation detector is used for detecting minute amounts of radioactivity, and is typically found in nuclear medicine facility. On the other hand, liquid scintillators are extensively used in research laboratories where beta-emitting radionuclides are primarily used. The Geiger-Muller counter is good for general surveying because of its high sensitivity. However, because of its avalanche effects, the Geiger-Muller counter is not used to measure radiation exposure. For accurate radiation measurements, ion chamber survey meters are generally used.

A typical Geiger-Muller counter has its reading expressed in mR/hr with different set scale factors: x0.1, x1, x10, and x100. The unit is equivalent to mrem/hr since they are used primarily to detect gamma particles and beta particles whose quality factor (w_R) is unity.

Figure 13.8 Geiger-Muller Counter

EXAMPLE 13.4 If the GM counter, shown in Figure 13.8, displayed a reading of 2.5, what is the measured exposure?

SOLUTION:

The radiation exposure X,

$$X = \text{Reading x Scale Factor}$$
$$= 2.5\,\text{mR/h} \times 10 = 25\ \text{mR/h}$$

The exposure measured is the value shown times the scale factor. The Geiger-Muller counter shown in Figure 13.8 is equipped to detect both photons and beta particles. The Geiger-Muller counter is equipped with a mechanism of displacing the metallic enclosure. This mechanism allows beta rays to pass through, with minimal attenuation, to the detector. Another design of ion chamber for the detection of beta particles is the pancake chamber that looks like a disk externally. The thin window also allows the detection of low energy radiation.

13.6 Personnel Monitoring

NCRP Report No. 33 recommends that personal monitoring be restricted to individuals that may receive a dose exceeding one-fourth of the maximum permissible dose.[16] **However, the NRC requires individuals to wear a monitoring device if the individual is likely to receive a radiation dose exceeding 10% of the equivalent dose limit in a year**.[17] Personnel monitoring helps to minimize health risks by regularly measuring individual radiation exposure and thereby disclosing improper radiation protection practices and potentially hazardous conditions. This leads to correction of the situation and adjustment of the work environment. In addition, personnel monitoring provides a method of documenting radiation exposures to individuals as well as spotting any sudden fluctuations of radiation exposure. Finally, personnel monitoring provides cumulative accounting of exposure allowing the identification of those individual exposures that may be near the effective dose limits.

Personnel monitoring devices include film badge, TLD badge, and pocket dosimeters. How these monitoring devices are being utilized depends on their monitoring functions. For whole body exposure monitoring, **body badges** are worn on the chest or abdominal region. For extremities or organ monitoring, the badges are placed at those particular areas. For example, **ring badges** are worn for monitoring exposure to the hands. Badges are placed either outside or under the apron. If placed beneath the apron, the exposure gives a true reading under the protective apron but not taking into account the exposure to the unprotected neck or head areas. On the other hand, if the badge is placed on the neck or taped outside the apron, the reading is much higher than that received beneath the apron. The utilization of the personnel dosimeters should be clearly identified and implemented consistently. Badges should be stored in a place away from radiation, excessive heat, water, humidity, and chemicals. Badges are returned at regular intervals, (usually monthly or quarterly), to the radiation safety office to be read and evaluated. If the exposure is excessive, the particular individual will be notified by the radiation safety office to resolve the situation.

Radiation monitoring badges are typically forwarded to a commercial company for evaluation from the radiation safety office. A report is provided and made available to the radiation worker. The evaluation is reported under DEEP (DDE), EYE (LDE), and SHALLOW (SDE) categories. The DEEP dose equivalent refers to the external whole-body exposure is measured at a tissue depth of 1 cm (1000 mg/cm^2). The EYE dose equivalent refers to the exposure of external lens of the eye is measured at a tissue depth of 0.3 cm (300 mg/cm^2). The SHALLOW dose equivalent refers to external exposure to the skin or an extremity and is measured at a tissue depth of 0.007 cm (7 mg/cm^2)

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[16] NCRP Report No. 33. *Medical X-ray and Gamma-ray Protection for Energies Up To 10 MeV – Equipment Design and Use*. Washington, (DC): National Council on Radiation Protection and Measurements; 1968. p. 33.

[17] NRC Regulations Part §20.1502. *Standards for protection against radiation – Conditions requiring individual monitoring of external and internal occupational dose.*

averaged over an area of one square centimeter. Dose equivalents arising from x-rays, gamma rays, and neutrons will have DEEP, EYE, and SHALLOW readouts. Dose equivalents arising from beta rays will only have SHALLOW readout. A result denoted by "M" indicates that the detected dose equivalent is below the minimal detectable level of the badge. The minimum detectable level is typically 10 mrem for a whole-body film badge and 20 – 30 mrem for the ring film badge. With the sensitivity of optically stimulated luminescence (OSL) technology, the minimum reported exposed level from Luxel dosimeters is now down to 1 mrem. For beta rays, the minimal detectable level is 40 mrem for both whole body and ring film badges.

It is still common to use **film badges** as personnel monitoring devices. In addition to the film, a film badge also has a multi-filter system consisting of an open window, aluminum, copper, lead/tin, and plastic filters. The filters are used to discriminate radiation of different energies, different types, and also to correct for energy dependence of the film. Film badges are considered very economical, costing about a dollar per month. Since film cannot be reused, it provides a legal permanent record. The disadvantages of film badges are that it takes time to process the film before the exposure is known, and the energy dependence of the film makes it less accurate compared to other detection systems. For the same amount of radiation exposure, a film exposed to 140 kVp photons would be darker compared to that exposed to gamma rays from a cobalt-60 machine. The interaction process is different because photoelectric effect is dominant at low photon energy while the Compton effect is the dominant at high photon energies as discussed in Chapter 11. To account for these different effects, filters of various materials such as aluminum and copper are placed over different portions of the film.

Thermoluminescence (TLD) badges are sometimes used for personnel monitoring. The advantage of TLD over film is its better low dose response, down to 5 mR, which results in less uncertainty; and the fact that it is not energy dependent. It is also less susceptible to water and hence, it is commonly used on ring badges. The TLD can be worn up to 90 days, although the general rule is to return the device for evaluation on a monthly basis. With its small size, TLDs are suitable for monitoring exposure at difficult to access areas, such as the rectum. The disadvantages of TLD are (a) it is more expensive than film and (b) it cannot be used, as a permanent record, since the TLD once read will have the number of trapped electrons returned to the valence band.

Pocket ionization dosimeters are used in cases where the workers are not routinely exposed to radiation. This is often the case for nursing care of patients undergone radioactive implants. The pocket dosimeter shown in Figure 13.9a is a mechanical type with pen-like external features and quartz-fiber electroscopes for measuring the exposure. Figure 13.9b shows a digital readout based on halogen quenched GM tube for the Victoreen pocket dosimeter.

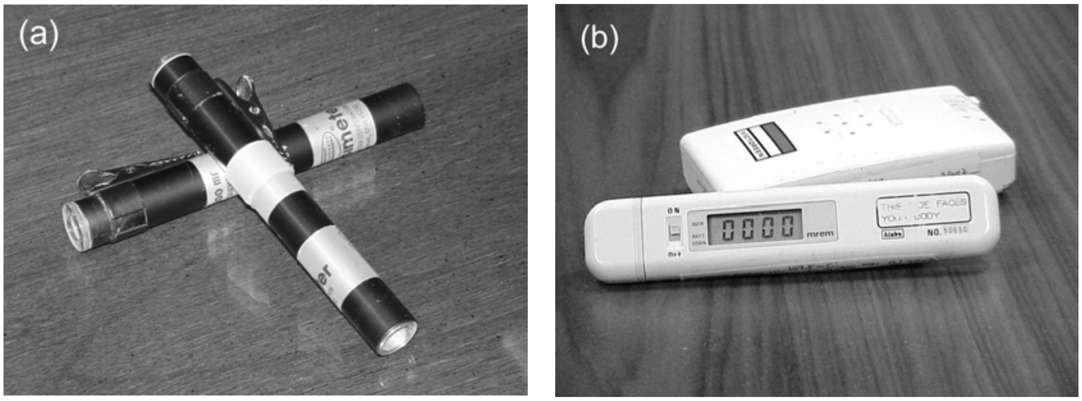

Figure 13.9 Pocket dosimeters: mechanical readout (left) and digital readout (right)

The mechanical type has a hermetically sealed ion chamber containing a small fiber electrometer for the detection of the presence of photons. Initially, the dosimeter is charged through a charge pin at one end of the dosimeter. At the other end, an optical eyepiece provides the viewing of the fiber line sets at zero scale when held up to the light. After exposure, the fiber line moves on the scale in mR or R by an amount proportional to the amount of charge discharged, which corresponds to the amount of accumulated exposure. The dosimeter is reset to zero when the electroscope is completely charged. The process of recharging takes only a few seconds and the dosimeters are ready for use again. Hence, pocket dosimeters are convenient but cannot be used as a permanent record. Like TLD, the readings from pocket dosimeters should be manually recorded before and after exposure.

13.7 Minimizing Exposure from External Sources

The amount of radiation exposure received by an individual and/or patient depends on the understanding of the effective methods of radiation protection even though there are regulatory guidelines for maximum exposure and warning signs of high radiation exposure. The three cardinal principles of radiation protection from external sources are **time, distance, and shielding**. It is advisable to complete the task as soon as possible to minimize the time of exposure. Radiation exposure is limited by maximizing the distance to the source and using shielding whenever possible. Sometimes, the implementation of one principle may interfere with another principle. For example, by wearing an apron, which is heavy and bulky, it would take longer to manipulate iridium-192 sources used in brachytherapy. Whether an apron should be worn under such conditions will depend on the assessment of the personnel to reduce the overall radiation exposure.

The amount of radiation exposure received by an individual is directly related to the duration of exposure. If the time of exposure is doubled, the amount of radiation to an individual is also doubled. The direct relationship of exposure to time can be written as

$$X = \dot{X} \times t \tag{13.5}$$

where X is the radiation exposure, $\dot{X}$ is the radiation exposure rate, and t is the exposed time, respectively.

EXAMPLE 13.5 A student was observing a brachytherapy procedure in a patient's room for about 15 min. If the exposure rate is 10 mrem per hour, what is his/her radiation exposure?
SOLUTION:
The radiation exposure X,

$$X = \dot{X} \times t$$

$$= \frac{10\,\text{mrem}}{1\,\text{hr}} \times 5\,\text{min} \times \frac{1\,\text{hr}}{60\,\text{min}} = \frac{5}{2}\,\text{mrem} = 2.5\ \text{mrem}$$

With this in mind, it is prudent to organize the tasks to be performed before entering a radiation area. The task should be executed as quickly as possible to reduce prolong exposure.

The radiation exposure decreases rapidly as the distance between the radiation source and the individual increases. If the distance is doubled, the radiation exposure is reduced to a quarter. This relationship is called the **inverse square law** effect and is exact if the radiating source is a point source. The inverse square law effect relating the exposure (X_1) at distance (d_1) to exposure (X_2) at distance (d_2) is given as

$$\frac{X_2}{X_1} = \left(\frac{d_1}{d_2}\right)^2 \tag{13.6}$$

where the subscripts 1 and 2 identify the exposures at the two distances.

EXAMPLE 13.6 In example 13.5, if the exposure rate is 2.5 mrem per hour at 1 meter, what is his/her exposure if the student stays at 3 meters from the sources?
SOLUTION:

$$X_2 = \left(\frac{d_1}{d_2}\right)^2 X_1$$

$$= \left(\frac{1}{3}\right)^2 2.5\,\text{mrem} = \frac{2.5}{9}\,\text{mrem} = 0.28\ \text{mrem}$$

Additional reduction in radiation exposure is possible if shielding is used between the source and the individual. The reduction in the amount of radiation exposure will depend on the type and thickness of the shielding materials as well as the type of radiation. The thickness of the shielding material is often described in terms of the half-value layer (HVL) or tenth-value layer (TVL). For shielding of radiation rooms, the thickness of materials is typically specified in terms of the TVL. One **TVL is the thickness of the material required to reduce the radiation intensity or exposure to one-tenth**

(or 10%) of its original value. On the other hand, **HVL is the thickness required to reduce the exposure to one-half (or 50%) of its original value**. One TVL is equal to 3.3 HVLs.

EXERCISE 13.2 Show that 1 TVL = 3.3 HVLs.

EXAMPLE 13.7 In EXAMPLE 13.6, if the student knows that the exposure is 0.28 mrem/hr and decides to stand behind a 5 HVL shield, what would be his/her exposure?
SOLUTION:

HVL	Transmission (%)
1	50
2	25
3	12.5
4	6.25
5	3.125

A simple transmission table based on the concept of half-value layers where the transmission intensity is reduced by half is constructed on the left side. After 5 HVLs, the intensity is reduced to 3.125%. In terms of exposure, this amounts to (3.125% x 0.28 mrem) 0.0088 mrem.

EXERCISE 13.3 Use the exponential equation $I/I_o = (1/2)^x$ to verify that the result in EXAMPLE 13.3 is correct.

A common shielding material is lead, which is effective in attenuating photons. Concrete is also used to shield photons. Borated materials are used to shield neutrons. The typical personnel shielding accessories in diagnostic radiology are **lead aprons**, **lead gloves** as well as **shielding plates** to eliminate unnecessary exposure to sensitive organs and other anatomical regions of interest. Lead aprons have shielding thickness equivalent to 1/4 or 1/2 mm Pb. The transmission of scattered radiation is about 10% or less depending on the energy of the primary radiation.

Figure 13.10 Bedside shield

In therapeutic radiology, **bedside shields** are used in patient's rooms to reduce exposure to staff personnel and visitors. These shields on wheels as shown in Figure 13.10 are about 1 inch thick. Different types of shields are used in hot lab to minimize exposure when preparing sources for patient loading or returning to the safe (Section 14.5). Likewise, shields are also used in the hot lab of nuclear medicine when radioisotopes are retrieved and prepared for patient administration. The word "shielding" has different interpretation when used in the context of radiation protection. In this context, shielding merely implies the reduction but not complete elimination of gamma radiation. Shielding cannot totally stop gamma radiation, but reduces it to a lower level of exposure.

13.8 Exposure from Sealed Sources

This section will focus on the aspects of radiation protection for patients receiving brachytherapy with sealed sources. Sealed sources that are currently used in brachytherapy are iodine-125, palladium-103, cesium-137, and iridium-192. The physical appearance of iodine-125 seeds, which is about 4 – 5 mm long and 0.8 mm in diameter, is shown in Figure 13.11. The physical sizes of palladium-103 and iridium-192 sources are about the same. On the other hand, the cesium-137 tube source commonly used in cervical applications is about 21 mm long and about 2.65 mm in diameter.[18] During a brachytherapy procedure the patient is implanted with radioactive sources. If the sources are inserted into tissues, the procedure is called **interstitial brachytherapy**. If the sources are placed inside the body cavity, the procedure is called **intracavitary brachytherapy**. If the source is placed on the surface facilitated with a mold, the procedure is called mold therapy or **superficial brachytherapy**. Brachytherapy procedures have also been classified according to the prescribed treatment time. If the sources are left permanently in the patient, the procedure is called **permanent implant.** Otherwise, the procedure is called **temporary implant**. Because these sources are sealed, there is generally no radiation contamination except for leakage.

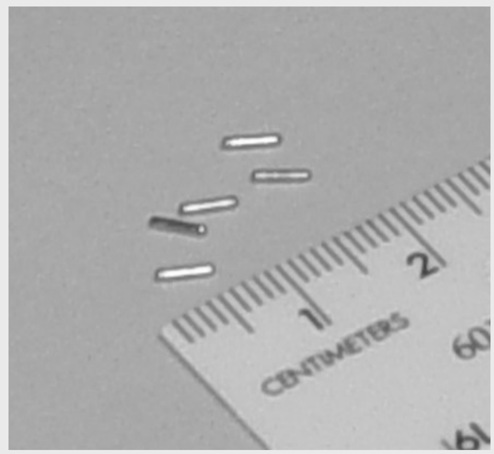

Figure 13.11 Iodine-125 seeds

Temporary implants are generally performed using afterloading techniques. In this technique, catheters, applicators, and/or needles are inserted near or into the tumor at the time of surgery. After surgery, the patient is brought to the radiation oncology facility to simulate the source positions using non-radioactive seeds (also called **dummy seeds**). After simulation, a dosimetric plan is generated and sources are prepared for loading into the patient. Depending on the exposure level, a patient receiving a temporary implant should be isolated and confined to a private room. During the implanting procedure, the sources implanted have to be counted, verified, and documented. Implantation should be performed using handling tools, which will increase the distance of personnel from the sources. If available, bedside shield (Figure 13.10) should be used to minimize radiation

[18] Glasgow, G.C; Perez, C.A. *Physics of Brachytherapy*. In: Perez, C.A.; Brady, L.W., editor. Principles and Practice of Radiation Oncology. 2nd Ed. Philadelphia: J.B. Lippincott Company; 1992. pp. 268-269.

exposure. After the implantation, the patient and the surrounding area have to be surveyed. Meanwhile, radioactive materials and radiation area warning signs should be placed at the entrance to the patient room. Radiation instructions should be written in the patient chart and the nursing staff should be informed that radiation precaution is in effect. During the treatment, a lead container called a **pig** and a handling tool such as forceps should be kept in the patient's room. If the radioactive sources become dislodged, these accessories will be used to pick up the sources and place them in the pig. It is bad practice to use bare hands to pick up radioactive sources. When the sources are removed at the prescribed time, the patient and the surrounding area should be surveyed and documented that all sources are accounted for. After the sources are removed, the radioactive materials and radiation area warning signs should be removed. In addition, a "radiation precaution discontinued" instruction should be written in the patient's chart and the nursing staff is informed of the status. All sources and associated handling tools are then returned to the hot room. The type and number of sources returned to the hot lab must be recorded in the logbook.

Permanent implants are usually performed in the operating room. The number of radioactive sources required for the implant is determined based on a pre-implant dosimetry technique. At the time of implantation, radioactive seeds are either injected or laid over the tumor or tumor bed. The patient, the staff, and surgical room are surveyed to ensure that there are no lost seeds after the implantation. In addition, the exposures at the surface and at 1 meter from the patient are measured. If the exposure is within regulatory guidelines, the patient is released immediately after recovery. At the time of release, the patient will be given instructions on how to deal with radiation precautions associated with the implants. If the exposure of the permanent implant exceeds the regulatory guidelines, the patient has to stay in the hospital until an acceptable exposure level is attained. Under such circumstances, the radiation precaution procedure should be followed just as the same for temporary implants.

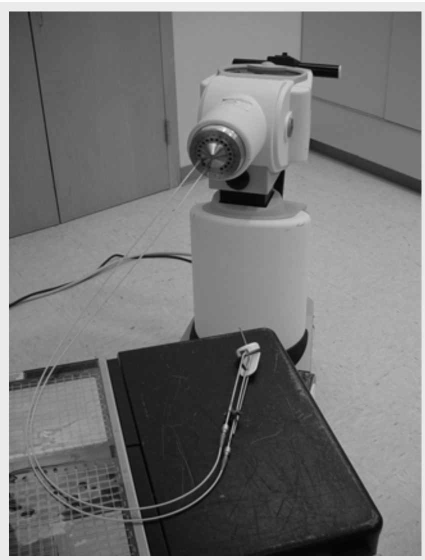

Figure 13.12 MicroSelectron HDR unit

High-dose-rate (HDR) remote after-loading brachytherapy is another type of temporary implant that uses a single high activity source to deliver the treatment dose to the tumor. The source strength of iridium-192 in the microSelectron HDR unit (shown in Figure 13.12) is about 10 Ci and has the dimension of about 0.7 mm in diameter and 4 mm long.[19] The source moves out of a special shielded storage container called a **safe** through transfer tubes to designated positions along the applicator. It

[19] Glasgow, G.C.; Anderson, L.L. *High dose rate remote afterloading equipment*. In: Nag S., editor. High Dose Rate Brachytherapy: A Textbook. Armonk, NY: Futura Publishing Company; 1994. p. 49.

dwells for a specified time at that position before moving to the next position. The procedure is repeated until the treatment is complete. After treatment, the source is retracted back into the safe. The movement of the source is controlled by a microprocessor. The treatment time is very short, in the order of minutes rather than hours. The treatment is typically performed on an outpatient basis in a shielded room. The procedure involves the initial insertion of catheters in the surgical room or examination room. After the catheters have been inserted, the positions of the sources are simulated and dosimetry performed. Before the patient can be treated, a quality assurance check is performed to verify the functionality of the system. During treatment, visual and audio systems are used for monitoring and communicating with the patient. After treatment, the patient and the surrounding areas are surveyed and documented that the source has returned to the safe.

The equation of exposure from a radioactive source is given as

$$X = \frac{A \cdot (\Gamma)_x \cdot t}{d^2} B \tag{13.7}$$

where X is the exposure, A is the amount of activity, $(\Gamma)_x$ is the exposure rate constant that relates activity to Roentgen for that radionuclide, t is the time being exposed, d is the distance from the source to the point where exposure is measured, and B is the attenuation (also called transmission) factor. It defines the fraction of beam intensity that is transmitted through the shielding material.

EXAMPLE 13.8 What is the exposure at 5 m from 8 seeds each containing 1 mCi of iodine-125 for two hours. The exposure rate constant for iodine-125 source is 1.45 $R \cdot cm^2 \cdot mCi^{-1} \cdot h^{-1}$
SOLUTION:

$$X = \frac{A\,\Gamma\,t}{d^2} B = \frac{(8 \text{ x } 1\,\text{mCi})(1.45\ \text{R} \cdot \text{cm}^2 \cdot \text{mCi}^{-1} \cdot \text{h}^{-1})(2\ \text{h})}{(5\ \text{m})^2}(1)$$

$$= \frac{23.2\ \text{R} \cdot \text{cm}^2}{25\ \text{m}^2} \text{ x } \frac{1\ \text{m}^2}{1 \text{ x } 10^4\ \text{cm}^2} \text{ x } \frac{1 \text{ x } 10^3\ \text{mR}}{1\ \text{R}}$$

$$= 9.28 \text{ x } 10^{-2}\ \text{mR}$$

13.9 Radioactive Contamination

Accidental **spillage of liquid radioactive materials** can result in the contamination of surrounding areas. The occurrence can happen in any place, in the preparation area, along the transporting route of radioactive sources, in the areas where patients undergo nuclear medicine scanning, or radiation treatment involving radioactive liquid or capsule, and in the laboratory area where radioactive materials are used. If contamination should occur, it would result in unwarranted radiation exposure. **To avoid the spread of contamination, use absorbent materials to absorb the spill**. **Soiled materials should be handled carefully with impermeable gloves and forceps**.

A designated commercial decontaminating solution or soap and water should be used to decontaminate the area. Check for signs of contamination on the shoes and if contaminated, they should be taken off and left in the contaminated area. If the area cannot be decontaminated or monitored immediately, the area should be covered using an impervious pad with the absorbent side down, and secured with a precautionary label of radioactive materials. **Isolate the contaminated area and detain all evacuees from the area until a qualified expert or personnel from the radiation safety office performs a radiation survey**.

If a spill on the skin occurs, the decontamination procedure should be enacted immediately to avoid radioactive material being absorbed into the skin. **All contaminated clothing should be removed**. If bleeding occurs, run water over the cut to allow the bleeding to continue and forcing out foreign materials. If the radioactive materials contain a chemical that is corrosive, follow neutralization procedures and then proceed with decontamination. Whenever lifesaving measures are needed, they should be performed first before proceeding with decontamination. **To perform skin decontamination, use soap and lukewarm water** or other designated commercial decontaminant solution **to wash off the affected area**. Notify the radiation safety office and physician administrating the radiation immediately.

Internal contamination of radioactive material is possible through skin penetration, ingestion, and inhalation. To avoid skin penetration **wear gloves when handling radioactive sources**. Properly discard worn gloves to avoid the spread of source contamination within the department or laboratory. When handling highly radioactive sources, it is better to wear two pairs of gloves.

To avoid ingestion, **do not eat, drink, or smoke in a laboratory and source handling area**. In addition, do not pipette radioactive solution by mouth. **Hands should be washed thoroughly after handling radioactive materials, and especially before eating**. It is always a good practice to keep the laboratory neat and clean to limit the spread of contamination.

Special absorbent padding with impervious backing should be available to cover work surface of the bench, thus limiting the spread of contamination in the event of an accident. The padding acts as a convenient means of removing spills without contaminating the surface of the benches or spreading contamination. Avoid the spread of contamination by wearing gloves and **using tongs when handling soiled materials such as bed sheets, cloths, and equipment**. Soiled materials should be **surveyed before disposal**. In addition, all areas where there is possibility of contamination such as the scanning area should be surveyed.

To avoid contamination via inhalation, **radioactive gases should be handled only under a hood**. When opening a vial containing radioactive iodine, the vial should be extended away from any person. In handling a therapeutic dose of iodine-131, it is better for the patient to open the vial and drink the dose by herself/himself. Although many iodine-131 therapy patients can be treated on an outpatient basis, some are confined to a private room until the radioactive materials decay or are eliminated naturally by the body, to an acceptable radiation exposure level. In this form of treatment, the patient can potentially cause significant contamination of items he/she touches. As such, an extensive room preparation and handling are associated with radioiodine treatment procedure. All items should be surveyed before being disposed.

13.10 Loss or Rupture of Sealed Sources

An appropriate plan should be implemented in the event a sealed source is misplaced. **The radiation safety officer (RSO) of the hospital, attending staff, or medical physicist involved should be informed immediately upon discovery**. A list of possible places where the source might have been transported to or carried through should be identified. A search for the source should be conducted using the most sensitive portable survey meter. Check the source transport route, patient room, and, if needed, the route where soiled materials are being transported including the laundry facilities. If the source is still not found, check the route of the trash to the incinerator. Check instrument trays if instruments are used and all areas including plumbing, fixture, and possible areas where it could have been dropped. The search must continue until the source is found or reported as being lost. All personnel involved may be interviewed in order to understand how the sources might have been misplaced. This is used to formulate corrective action if necessary.

Due to rough handling, high temperature, or crushing, it is possible that **a sealed source can be ruptured**. Depending on the type of source, it may be toxic. Regardless of the nature of the source, the area should be closed off immediately and the source should be placed in a sealable container if possible. If it is airborne, the air ventilation should be shutdown. Place a damp cloth over the suspected source. Personnel movement should be limited to avoid any spread of radioactive contamination. Do not dispose of any material. Inform the RSO for further actions regarding the removal of the source and the decontamination of the area. Decontamination should include a wipe test of the involved area.

Iodine-125 seeds have been reported to leak. This leakage may be due to physical damage by the mishandling of forceps and scissors. As a result of the damage, iodine-125 has been found in the patient's urine and thyroid, and exposure to personnel handling the source. The amount of contamination can be determined using thyroid uptake probes and whole body imaging. An uptake probe is basically a counter as shown in Figure 13.13. In addition to damage, iodine-125 had been known to dislodge from the prostate and found in the urine specimen collected a few days after implantation. Although the sources may be flushed into the toilet to minimize exposure, it is more appropriate to follow the instructions given by the radiation oncology facility that performed the implant at the time of discharge.

13.11 Instructions to Allied Medical Workers


Figure 13.13 An uptake probe

Generally, the potential radiation exposure to allied medical workers is low. Still, radiation exposure should be kept to a minimum following the ALARA principle. As such, **allied medical workers should be made aware of the location of restricted areas**. Allied medical workers such as housekeeping, security, and maintenance personnel should **obtain permission and instructions from the person in charge or RSO before entering the restricted area**. Likewise, any work that has to be performed or any item that has to be removed from the restricted area should be carried out with the consent of the person in charge or RSO. **Housekeeping personnel should not remove items from rooms posted with radioactive material signs**. Security personnel should be aware of locations where packages of radioactive materials are placed. **Allied personnel should be trained to refrain from eating, smoking, or drinking in areas where radioactive materials are used**. The general guides for the safe handling of radioactive materials are given in Table 13.5.

Table 13.5 Guidelines for the safe handling of radioactive materials

1. Preparation of radioactive materials must be carried out in designated areas such as the hot lab, supervised by an instructor.
2. Radioactive materials are to be issued to authorized persons. All radioactive materials must be returned and accounted for.
3. Gloves should be worn when handling radioactive liquid.
4. Eating, drinking or using cosmetics are not permitted in the hot lab.
5. Pipetting must never be done by mouth. Use suction devices designed for pipettes. Contaminated suction devices should be discarded.
6. All persons working with radioactive materials should be familiar with the operation of the radiation survey instruments available in the hot lab.
7. A survey should be promptly performed upon suspicion that an area, clothing, or hands have become contaminated.
8. All exempted radioactive waste should be labeled and discarded after 10 half-lives.
9. Any cut or wound arising from working with radioactive materials should be immediately reported and cleaned.
10. Persons working with liquid sources should wash their hands and survey their person for contamination prior to leaving the hot lab.

13.12 Radiation Emergencies

Radiation emergency refers to any catastrophic event involving radiation exposure to any person requiring medical care. The emergency can be dramatic, such as exposure from a nuclear accident, or overexposure in a laboratory. Each facility should have its own emergency procedure for possible events that may occur near the facility. It involves coordination of the appropriate personnel and procedures on how to resolve the medical condition under contaminated environment.[20] Such procedures should be brought to the attention of the allied medical workers at the time of orientation and during radiation safety training sessions. Consult the RSO for explicit instructions on how to proceed from a radiation environment in unforeseen events such as fire, flood, and explosion.

Summary

13.1 Different types of radiation produce different radiobiological effect.

13.2 Equivalent dose is used principally in radiation protection to express the same radiobiological effect from all types of radiation. The total equivalent dose is the sum of the product of the absorbed dose and the weighting factor (w_R) of each type of radiation.

13.3 Environmental radiation consists of background radiation and radiation from man-made sources. The three sources of background radiation are cosmic rays, terrestrial radiation, and internal radiation. They contribute approximately 1 mSv per year excluding radon, which contributes 2 mSv per year in the United States. The largest radiation exposure from man-made sources is from medical procedures.

13.4 Advisory bodies like the ICRP and NCRP have been formed to make recommendations on the use of radiation. These advisory bodies do not establish regulations. The NRC and the individual agreement states establish regulations for the safe use of radiation.

13.5 The effective dose equivalent concept was introduced to address the issue of non-uniform body irradiation. The effective dose equivalent is the sum of the weighted dose equivalent for the irradiated tissue or organs. It takes into account the different mortality risks from cancer and the risk of severe hereditary effects in the first two generations associated with irradiation of different organs and tissues. These weighting factors are listed in NCRP Report No. 91.

13.6 The cardinal rules of minimizing radiation exposure from external sources are whenever possible, (a) minimize the time spent near the source, (b) maximize the distance away from the source, and (c) place shielding so that there is no direct exposure from the source.

13.7 For radiation protection purpose, the typical radiation detecting instruments are film, TLD, GM counters, ionization chambers, scintillation counters, and solid-state diodes.

13.8 The typical personnel monitoring systems include film badges, TLD badges, OSL badges, and pocket dosimeters. Film badges, TLD and OSL badges are routinely used to monitor exposures, which are read monthly or quarterly. On the other hand, exposed pocket dosimeters can be read and reset immediately. As such, pocket dosimeters are used whenever there is a need for immediate exposure readings or where routine monitoring is not implemented.

[20] NCRP Report No. 111. *Developing Radiation Emergency Plans For Academic, Medical or Industrial Facilities*. Bethesda, (MD): National Council on Radiation Protection and Measurements; 1991.

13.9 A cylindrical Geiger-Muller counter has a metallic enclosure displacement mechanism to allow the beta rays to enter the detector. A pancake Geiger-Muller detector has a large volume with thin window for detecting small amounts of low energy beta particles.

13.10 Like the Geiger-Muller pancake chambers, scintillation counters are used to detect minute amounts of low energy radiation.

13.11 In radiation oncology, radioactive seed sources should be handled using forceps and manipulation of sources should be carried out behind a shield. Sources should be transported in a shield or pig, and along a path that minimizes exposure to the public. The patient and the room should be surveyed after implantation and after removal of sources from the patient.

13.12 Gloves should be worn for handling liquid radioactive sources. Again, handling of liquid radioactive sources should be carried out behind a shield. Sources should be transported in a shielded container. Areas where radioactive materials are administered should be surveyed on a regular basis.

13.13 A summary of formulas:

Equivalent dose: $H_T = \sum(w_R \times D_{T,R})$

Effective dose: $E = \sum(w_T \times H_T)$

Film optical density: $OD = \log_{10}\left(\dfrac{I_o}{I_t}\right)$

Exposure from a radioactive source: $X = \dfrac{A\,(\Gamma)_x\, t}{d^2}\, B$

Study Guide

13.1 In your own words, define the following terms:

(a) equivalent dose	(b) radiation weighting factor (w_R)
(c) background radiation	(d) NCRP
(e) NRC	(f) ALARA
(g) effective dose limit	(h) MPD
(i) agreement states	(j) tissue or organ weighting factor
(k) byproduct materials	(l) saturation in ion chamber
(m) personal monitoring	(n) areal monitoring
(o) HVL	(p) TVL

13.2 Identify the rationale for the introduction of the equivalent dose concept.

13.3 List the radiation weighting factors for x-rays, gamma rays, and beta rays.

13.4 List the three sources of background radiation.

13.5 Excluding radon, what is the annual background radiation? Express your answer in rems.

13.6 What is the estimated annual exposure to the population from (a) medically used man-made sources and (b) radon?

13.7 What are the roles of NCRP, ICRP, and NRC in the medical use of radiation?

13.8 Identify any three agreement states. Is your state an agreement state?

13.9 Which agencies regulate the safe use of (a) cobalt-60, (b) iodine-131, (c) medical linear accelerators, (d) x-ray equipment, and (e) particle accelerators?

13.10 Give various situations where the ALARA principles may be implemented.

13.11 List the effective dose limits recommended by NCRP for (a) occupational worker, (b) the public, (c) a pregnant worker, (d) the eyes, and (e) the extremities of a radiation worker.

13.12 A person receives iodine-131 thyroid therapy and is then released. Should the exposure from this medical procedure be included in the dose limit set for the general public?

13.13 Explain maximum permissible dose. On what basis was it derived?

13.14 Identify the bias voltage range for each type of radiation detectors: (a) diodes, (b) ionization chambers, and (c) Geiger-Muller counters.

13.15 Explain the principle of the operation of a Geiger-Muller counter. Discuss its advantages and disadvantages as a survey instrument.

13.16 Your GM counter was reading 1.5 with a setting on the x0.1 scale. What is your measured exposure rate?

13.17 What is the highest scale on your GM counter?

13.18 Explain why a liquid scintillation counter is appropriate for the detection of tritium.

13.19 List three types of personnel radiation monitoring devices.

13.20 What criteria should be used to determine whether a person working in a controlled area should wear a radiation badge?

13.21 Explain the operation of each detection system: (a) film badge, (b) TLD badge, and (c) pocket dosimeter.

13.22 List the advantages and disadvantages of each of the following radiation monitoring systems: (a) film badge, (b) TLD badge, and (c) pocket dosimeter.

13.23 Discuss why a body badge should be worn under an apron. At which anatomical location should you wear your badge?

13.24 Under what circumstances should an individual wear two badges?

13.25 Determine whether wearing a 0.5 mm Pb equivalent apron is an effective means of radiation protection under the following conditions:
(a) an iridium-192 implant (b) an iodine-125 implant
(c) a diagnostic x-ray procedure (d) a technetium-99 patient
(e) positron scanning procedure

13.26 One TVL is equivalent to how many HVLs?

13.27 Explain why it is a good practice not to use bare hands to hold radioactive sources?

*13.28 Explain why people living in the polar regions receive more cosmic rays than those living in the equatorial region?

*13.29 How does the plastic CR39 Fast Neutron Dosimeter work?

Problems

13.1 What is the total equivalent dose if a person is exposed to 1000 rads of thermal neutrons and 500 rads of electrons? Express the result in Sv.

13.2 Background radiation exposure is about 1 mSv per year. Express this value in units of (a) rem and (b) mrem.

13.3 Express 1 mrem in μSv.

13.4 The exposure limit to a declared pregnant worker is 5 mSv. Convert this value to mrem.

13.5 What is the total effective dose received by a 40 year old worker who is exposed to the dose limit annually since the age of 27? How does this dose compare to the cumulative dose according to NCRP Report No. 91?

13.6 After a film has been exposed to 150 cGy and processed, the optical density was found to be 2.0. What is the transmitted intensity compared with the original intensity as measured using a densitometer?

13.7 A student watches a brachytherapy procedure in a patient's room. If the student is 2 m away and receives 6 mR/hr, what would be the exposure rate if the student moves to a position 3 m away from the patient?

13.8 A radiation oncology resident responded to the concerns of a brachytherapy patient. If it took 20 minutes to satisfactorily answer the patient's concerns, what was the exposure to the resident if the exposure rate was 15 mR/hr?

13.9 The probability that a monoenergetic photon beam will be attenuated by a 2 mm thick lead is 75%. What is the HVL of this beam?

13.10 A line ribbon consisting of 10 iridium-192 seeds was dropped during a procedure. What was the exposure rate in mR/hr at the patient's room door 4 m away? Assume that the activity of each seed was 1 mCi and the exposure rate constant for Ir-192 is 4.6 $R \cdot cm^2/(mCi \cdot hr)$. Is this amount of activity detectable at this location?

*13.11 What is the initial exposure rate of chromium-51 if it is to yield 0.5 R at 1 m for a complete decay? What is the corresponding activity assuming that the exposure rate constant of chromium-51 is 0.015 $mR \cdot m^2 \cdot mCi^{-1} \cdot hr^{-1}$?

*13.12 Since the age of 18, a radiation worker was exposed annually to the effective dose limit. At what age would the annual effective dose limit exceed the cumulative dose limit by 100 mSv?

*13.13 Work through this problem by following NCRP Report No. 39. A 30-year-old radiation worker has an exposure history of 61 rem. He received an additional 2 rem this year at work, followed by 3 rems exposure from a barium enema study to diagnose his medical problem. Calculate the maximum exposure he can receive for the remainder of this year (end of his 31st birthday).

Multiple Choice Questions

Select the one correct answer.

13.1 The dose equivalent is ___ times greater than the absorbed dose for 10 MeV neutrons.
a) 1
b) 5
c) 10
d) 20
e) none of the above

13.2 Which of the following is NOT TRUE of the radiation weighting factors?
a) The radiation weighting factors for x-rays, gamma rays, and electrons are identical.
b) The radiation weighting factor is 1 for x-rays.
c) The radiation weighting factor is a dimensionless quantity.
d) The radiation weighting factor accounts for the destructiveness of different radiation types.
e) none of the above.

13.3 Excluding radon, the average annual background radiations is
a) 1 mSv.
b) 2 mSv.
c) 3 mSv.
d) 4 mSv.
e) none of the above.

13.4 The effective dose limit for a radiation oncologist is
a) 1 mSv/yr.
b) 5 mSv/yr.
c) 15 mSv/yr.
d) 50 mSv/yr.
e) none of the above.

13.5 The maximum allowable radiation exposure to an unbadged employee of a radiation therapy facility is
a) 100 mrem per year.
b) 10 mrem per week.
c) 1 Sv per year.
d) 10 mSv per year.
e) none of the above.

13.6 Which of the following is NOT TRUE of the NRC?
a) The NRC regulates the use of radium.
b) The NRC regulates all reactor-produced byproducts.
c) The NRC is formerly called the atomic energy commission.
d) Agreement states also assume the role of the NRC.
e) none of the above.

13.7 Which of the following is NOT TRUE of the film badge?
a) It serves as a permanent record.
b) It is cheaper than TLDs.
c) The low dose response is better than TLDs.
d) It is susceptible to water.
e) none of the above.

13.8 The amplitude of the signal from a proportional counter
a) is the same for all types of radiation.
b) is proportional to the number of ions produced by a charged particle traversing the chamber.
c) will be greater for an electron than for an alpha particle.
d) is proportional to the exposure rate.
e) none of the above.

13.9 Which of the following is not the cardinal principle of minimizing radiation exposure?
a) Time
b) Distance
c) Shielding
d) Education
e) none of the above

13.10 The primary purpose of the filter in a film badge is
a) to absorb scattered radiation.
b) to secure the optical density range.
c) to allow the measurement of dose over a range of energies.
d) to change the slope of the H&D curve.
e) none of the above.