13 Evaluation of Radiation Safety Measures

The effectiveness of safety measures against radiation hazards is evaluated by a surveillance program that includes the observation of both people and their environments. Such a surveillance program may employ one or more of a variety of techniques, depending on the nature of the hazard and the consequences of a breakdown in the system of controls. These techniques may include preemployment and periodic physical examinations, estimation of internally deposited radioactivity by bioassay, and total-body counting, personnel monitoring, radiation and contamination surveys, and continuous environmental monitoring.

MEDICAL SURVEILLANCE

The great degree of overexposure required before clinical signs or symptoms of overexposure appear precludes the use of medical surveillance of radiation workers as a routine monitoring device. Nevertheless, medical supervision may play an important role in protecting radiation workers against possible radiation damage. Among the main tasks of medical supervision is the proper placement of radiation workers according to their medical histories, physical condition, and history of previous radiation exposure. Dermatitis, cataracts, and blood dyscrasias, including leukemia, are effects associated with radiation exposure. A preemployment physical examination, therefore, should be given if the nature of the work, including consideration of possible accidental overexposures, warrants it. Special attention is paid to physical conditions including fitness for use of respirators that may lead to, or be suggestive of, susceptibility to any of these effects. Possible indirect effects from working with radioisotopes are also considered by the examining physician. For example, sensitivity or allergy may contraindicate work that requires the wearing of rubber gloves or that may require washing the hands or body with strong detergents or harsh chemicals in order to decontaminate the skin. In addition to the preemployment examination, the radiation

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worker may be routinely examined at periodic intervals to ascertain that he or she continues to be free of signs that would contraindicate further occupational exposure to radiation. The physician is thus instrumental in preventing damage or injury that could otherwise arise, either directly or indirectly, as a consequence of working with radioisotopes or exposure to radiation. Medical supervision of radiation workers may also be necessary to evaluate overexposure, to treat radiation injuries, and to decontaminate personnel. These activities of the physician are, of course, in addition to the provision of routine health services that are not connected to radiation hazards. It should be pointed out that medical surveillance of workers is not unique to the field of radiation health. All good occupational health programs include preemployment examinations, consideration of medical findings in job placement, and continuing medical surveillance to help maximize the protection of workers against the harmful effects of toxic substances.

ESTIMATION OF INTERNALLY DEPOSITED RADIOACTIVITY

One of the techniques for estimating the intake of radioactivity or the radiation dose from that intake in order to demonstrate compliance with regulatory requirements for limiting the total effective dose equivalent (TEDE) is by a program of routine bioassay. Since the likelihood of intake is related to the level of environmental contamination, routine bioassay programs are designed to verify the efficacy of environmental contamination controls. In routine bioassay, the worker's body burden of radioactivity is determined periodically on a fixed frequency. The results are then compared to an arbitrarily established reference level that is not expected to be reached if there is no breakdown in contamination control measures. The method and frequency of routine bioassay monitoring is determined by the radiation characteristics and metabolic kinetics of the radionuclides of interest as well as by the sensitivity of the monitoring methods, the acceptable degree of uncertainty of the implied dose, and the ease and convenience of the method.

In addition to routine bioassay, special bioassay determinations are made in cases of suspected or known accidental exposures by inhalation, ingestion, or through wounds. When used as a monitoring technique, an administrative level, called an *investigation level*, is set. An investigation level is an estimated intake or committed dose that is higher than expected and above which the result is considered sufficiently significant to warrant further investigation.

Bioassay programs rely on two general techniques. One of these techniques is called in-vivo bioassay; it is the direct determination of internal radionuclides by whole-body counting and is useful only for gamma-emitting radionuclides or beta emitters that give rise to suitable bremsstrahlung. The other technique is called in-vitro bioassay, and involves the analysis of body fluids, exhaled air, or excreta for the purpose of estimating the intake.

Bioassay

The IAEA (1996)[1] and the ICRU (2002)[2] both provide guidance on in-vivo measurements of radionuclides. ICRP 130 makes multiple recommendations on in-vivo and

[1]IAEA, 1996. Basic Safety Standards for Direct Methods for Measuring Radionuclides in the Human Body. Safety Series 114. International Atomic Energy Agency, Vienna.
[2]ICRU, 2002. Retrospective assessment of exposure to ionising radiation. ICRU Report 67. *J. ICRU.* 2(2).

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in-vitro bioassay. The four categories of bioassay monitoring programs delineated by the ICRP are:

1. *Routine monitoring*, where workers may possibly be exposed during normal operations
2. *Special monitoring*, typically after an event or suspected event.
3. *Confirmatory monitoring*, to verify routine monitoring is not required.
4. *Task-related monitoring*, when a specific task(s) might result in an intake.

Monitoring is initiated after derived investigation levels (DILs) have been exceeded. A DIL can be directly converted to an investigation level (IL). An IL is typically set by a facility to correspond to an effective dose. Besides whole-body counting, urine and fecal samples may be used for bioassay. Nasal smears and "nose blow" are more of a screening technique.

In-Vitro Bioassay

The underlying rationale for in-vitro bioassay is that a quantitative relationship exists among inhalation or ingestion of a radionuclide, the resulting body burden, and the rate at which the radionuclide is eliminated. From measurements of activity in the urine or feces, therefore, we should be able to infer the body burden, and from the body burden, we can estimate the resulting dose. Unfortunately, the kinetics of metabolism of most substances in any particular individual is influenced by a large number of factors; as a consequence, there is a great deal of uncertainty about the exact quantitative relationships among elimination rates, body burden, and radiation dose. In most instances, therefore, bioassay data give only a very approximate estimate of the intake and dose. Although both urine and feces are available for bioassay measurements in case of an accidental inhalation or ingestion of a large amount of radioactivity, routine bioassay monitoring is usually done with urine samples, because of the ease of sample collection and also for esthetic reasons.

Readily soluble radionuclides may be grouped into three categories according to their metabolic pathways and distribution within the body: (1) those that are uniformly distributed throughout the body, such as ^{3}H in tritiated water or radiosodium ions; (2) those that concentrate mainly either in specific organs, such as iodine in the thyroid gland or mercury in the kidney; and (3) those that are deposited in the skeleton. Bioassays are most reliable in the case of the first category, the widely distributed radionuclides. In this case, the radioactivity in the body decreases exponentially at a rate given by the effective elimination constant λ_E and the body burden $A(t)$ at any time t after an intake $A(0)$ given by

$$A(t) = A(0)e^{-\lambda_E t}. \tag{13.1}$$

If a constant fraction of the isotope f_U is eliminated in the urine, then the activity in the urine $U(t)$ at time t after ingestion is given by

$$U(t) = f_U \frac{dA(t)}{dt} \tag{13.2}$$

and

$$U(t) = f_U A(0)\lambda_E e^{\lambda_E t}. \tag{13.3}$$

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TABLE 13-1 Water Balance for Reference People (ICRP 89 and ICRP 23)

	ICRP 89 Male	ICRP 89 Female	ICRP 23 Male	ICRP 23 Female	ICRP 23 10 yrs
Water intake in food and fluids (mL/d)	2600	1960	3000	2100	2000
Oxidation of food (mL/d)	300	225	350	250	200
Losses (mL/d)					
Urine	1600	1200	1400	1000	1000
Feces	110	95	100	80	70
Insensible loss	690	515	850	600	580
Sweat	500	375	650	420	350

Sources: Basic Anatomical and Physiological Data for Use in Radiological Protection Reference Values. ICRP Publication 89. *Ann. ICRP*, 32(3–4). ICRP 1975, *Report on the Task Group on Reference Man*. ICRP Publication 23. *Ann. ICRP.*

The principles of internal exposure evaluation may be illustrated with an example using tritium as tritiated water. Tritiated water taken in by a reference person either by inhalation, ingestion, or through the intact skin is completely absorbed and instantaneously and uniformly distributed throughout the body fluids in the soft tissues. In this case, the tritium concentration in the urine is the same as that in the body fluids and, hence, serves as a measure of the ^{3}H body burden.

The reference adult male body contains 42 L of water. Three liters of water are turned over per day through the ordinary excretion pathways according to ICRP 23 (Table 13-1). The daily turnover rate of body water λ_B in the ICRP 23 reference adult male is

$$\lambda_B = \frac{3\frac{\text{L}}{\text{d}}}{42\ \text{L}} = 0.071\ \text{per day},$$

which corresponds to a biological half-time T_B of 9.7 days. Of the 3 L of water lost per day, 1.4 L, or 47%, is excreted in the urine. Studies on humans found that the retention of ^{3}H following an intake of tritiated water is described mathematically (ICRP 30 and 54) by a three-component function:

$$R(t) = Ae^{-\frac{0.693t}{T_1}} + Be^{-\frac{0.693t}{T_2}} + Ce^{-\frac{0.693t}{T_3}}, \tag{13.4}$$

where T_1 ranges from 6 to 18 days, T_2 from 21 to 34 days, and T_3 from 250 to 550 days. The last two components represent tritium in organic compounds and together contribute no more than 10% of the absorbed dose. For this reason, we assume that the retention of ^{3}H taken in as tritiated water is adequately described by a single-component exponential equation with a half-time of 10 days. If the activity of the intake is $A(0)$, then the activity retained after a time t days following the intake is

$$R(t) = A(0)e^{\frac{-0.693t}{10}} = A(0)e^{-0.069t}, \tag{13.5}$$

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and the intake retention fraction, IRF (as computed by the intake retention function) is

$$\mathrm{IRF}(t) = \frac{R(t)}{A(0)} = e^{-0.069t}. \tag{13.6}$$

Since the specific activity of the urine is the same as the specific activity of the tritiated water in the body, and since the water volume of the reference adult is 42,000 mL, then the expected specific activity C_U of the urine at time t days after a single intake is

$$C_U(t) = \frac{A_0}{42{,}000\ \mathrm{mL}} \cdot e^{-0.069t}. \tag{13.7}$$

The calculated dose commitment based on urinalyses over a 7-day period differs from that estimated from just a single urine sample. The difference in this case is due to the fact that the mechanic's tritium clearance rate differed significantly from the clearance rate on which the ICRP model is based. In case of a real or suspected overexposure, it is recommended that as many real data as are available be used to estimate intake or to calculate the dose from the estimated intake. Generally, if an accidental intake estimate that is based on ICRP IRFs shows the intake to have been much less than the ALI, no further attempts at increased accuracy are necessary. However, if the estimated intake is significant relative to the ALI, then additional data should be obtained in order to determine the intake and the resultant dose as accurately as possible. Such additional data may include further bioassay data from sequential sampling, as in the example above, breathing-zone atmospheric concentrations from air sampling, and whole-body counting if applicable.

For the second category, those radionuclides that concentrate in one or more organs, there is a greater degree of uncertainty in the intakes and doses estimated from bioassay monitoring than in the case of the widely distributed radionuclides. These radionuclides are absorbed, after ingestion or inhalation, into the body fluids and the blood plasma. From these fluids, they pass into the organs in which they concentrate; a dynamic equilibrium eventually results between the concentration of the nuclide in the organ and that in the body fluids. While the isotope is equilibrating between the body fluids and the organ of concentration, it is also being filtered by the kidney into the urine. This leads to a clearance curve (Fig. 13-2) that is the sum of at least two exponential components. The first component, which falls steeply, represents the clearance of the isotope from the body fluids, while the

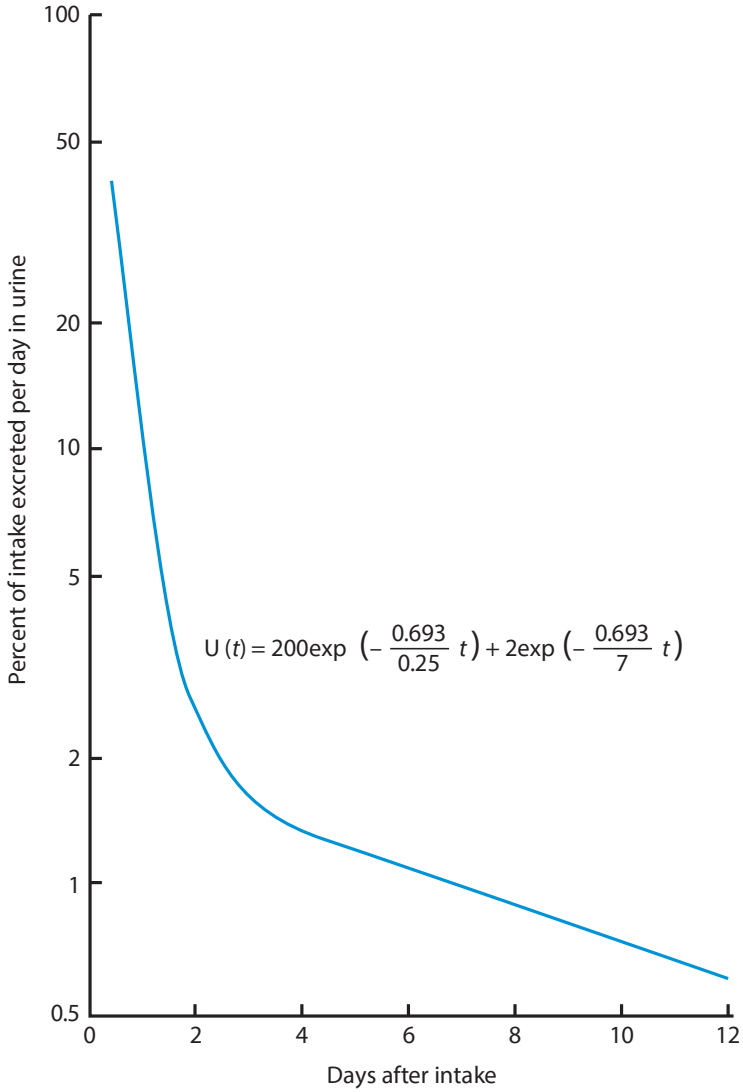

Figure 13-2. Urinary excretion curve of ^{35}S after a single intake of soluble, inorganic sulfate. The general shape of the curve is typical for a readily soluble compound that concentrates in a single organ. (Reproduced with permission from Jackson S, Dolphin GW. Report AHSB (RP) R 51. UK: Atomic Energy Authority (UK AEA); 1965.)

second component represents the clearance of the isotope from the organ of concentration. The slope and magnitude of the first component may be influenced by a number of factors, such as the amount of the nonradioactive form of the same element that was inhaled or ingested, the amount of water intake, the physiologic state of the kidney, etc., which makes it extremely difficult to relate the intake of the radionuclide to the urinary excretion data during the first few days after a single intake. The component that represents the clearance from the organ of concentration is much less influenced by these factors. An approximate estimate, therefore, of the radionuclide in the organ of concentration following a single exposure can often be made from the urinary excretion after sufficient data become available to establish the second component of the curve.

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The third category, which comprises the elements absorbed into the bone, is a special case of the category of isotopes concentrated in an organ or tissue. Bone seekers differ from other radionuclides mainly in the elimination of the isotope. While clearance half-times for non–bone seekers are measured in days or weeks, retention times for the bone seekers are measured in years. Furthermore, the clearance rate from the skeleton is not constant but decreases with increasing time. This is due to the fact that the skeleton is not a single "compartment" but rather a number of different compartments, each of which has its own clearance rate. Over a long period of time, the IRF, which is the sum of all the exponentials representing these different compartments, can be approximated by a power function of the form

$$R(t) = At^{-n}, \tag{13.8}$$

where

$R(t)$ = fractional retention t days after intake,
A = normalized fraction of the intake retained at the end of 1 day, and
n = an empirical constant.

For the case of ^{226}Ra, for example, $A = 0.54$, and $n = 0.52$.

Later studies on radium retention were able to resolve the bone into five compartments, (ICRP 54) and the following retention equation, which gives the activity retained per unit activity uptake at time t days after the uptake, was fitted to the observational data:

$$R(t) = 0.54e^{-1.73t} + 0.29e^{-0.139t} + 0.11e^{-0.116t} + 0.04e^{-9.9\times10^{-4}t} + 0.02e^{-1.39\times10^{-4}t}. \tag{13.9}$$

The corresponding urinary excretion curve, expressed as the fraction of the uptake activity per 24-hour urine sample at time t days after the uptake, is

$$f_U = 0.047e^{-1.73t} + 0.002e^{-0.139t} + 6.6\times10^{-5}e^{-0.116t} + 2\times10^{-6}e^{-9.9\times10^{-4}t} + 1.4\times10^{-7}e^{-1.39\times10^{-4}t}. \tag{13.10}$$

The body burden of radium may also be inferred from measurements of radon concentration in the breath. Radium transforms directly into radon; some of the radon dissolves in the body fluids and in the adipose tissue, and the balance is exhaled. Although the fractional retention of radon in the body varies for a short time after deposition of the radium, the mean exhalation is given as 70%. The exhaled radon activity, A_e, is related to the body burden q by the equation

$$A_e \text{ Bq/min} = 0.7 \cdot q \text{ Bq} \cdot \lambda \text{ min}^{-1}, \tag{13.11}$$

where λ, the decay constant for radium, is 8.1×10^{-10} min^{-1}. The concentration of radon in the breath C_B for a volumetric respiration rate of V L/min is given by

$$C_B \frac{\text{Bq}}{\text{L}} = \frac{A_e \dfrac{\text{Bq}}{\text{min}}}{V \dfrac{\text{L}}{\text{min}}}. \tag{13.12}$$

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Under resting conditions, the respiration rate is about 20 per minute and the tidal volume is about 0.5 L; the ventilation rate V, therefore, is about 10 L/min. For analysis, radon from a measured volume of exhaled breath is adsorbed on activated charcoal. It is then desorbed and transferred into an ionization chamber or scintillation cell for counting.

For strontium, another bone seeker, the biological retention function, after uptake of a unit of Sr, is given (ICRP 54) by the sum of three exponentials, with biological half-times of 3 days, 44 days, and 4000 days (11 years):

$$R(t) = 0.73e^{-\frac{0.693}{3}\cdot t} + 0.1e^{-\frac{0.693}{44}\cdot t} + 0.17e^{-\frac{0.693}{4000}\cdot t}, \tag{13.13}$$

and the urinary excretion function, f_U, expressed as the fraction of the uptake strontium excreted in a 24-hour urine sample t days after the uptake, is

$$f_U(t) = 0.13e^{-\frac{0.693}{3}\cdot t} + 0.001e^{-\frac{0.693}{44}\cdot t} + 2.4\times10^{-5}e^{-\frac{0.693}{4000}\cdot t}. \tag{13.14}$$

Using the principles described above, together with data from physiologically based biokinetic models applied to the reference man, tables of IRFs and urinary and fecal elimination fractions for most radionuclides that were ingested or inhaled were calculated and together with the calculational methodology for their application are published in Lessard (1987)[4] and in ICRP 78.

In-Vivo Bioassay

In-vivo bioassay may range from simply measuring the radiation from the thyroid gland under standardized conditions with a simple detector such as the Geiger counter probe, with a scintillation detector from a health physics surveying instrument, or with a highly sophisticated, heavily shielded, low-background whole-body counter. The simple determination may be used in routine "go–no go" monitoring to determine whether a previously established reference level has been reached, while measurements with the sophisticated system are useful to learn how much and what kind of activity is in the body. A whole-body count directly measures the radiation emitted from the internally deposited radionuclides. This information is used to determine the nature and location of the radionuclide and to quantitatively estimate the amount in the body, as shown in Figure 13-3. Generally, gamma emitters whose quantum energy exceeds about 30–50 keV can be determined using in vivo techniques. When in vivo methods are used to determine internal emitters, the in vivo data are customarily used as the data of record for demonstrating compliance with radiation safety regulations.

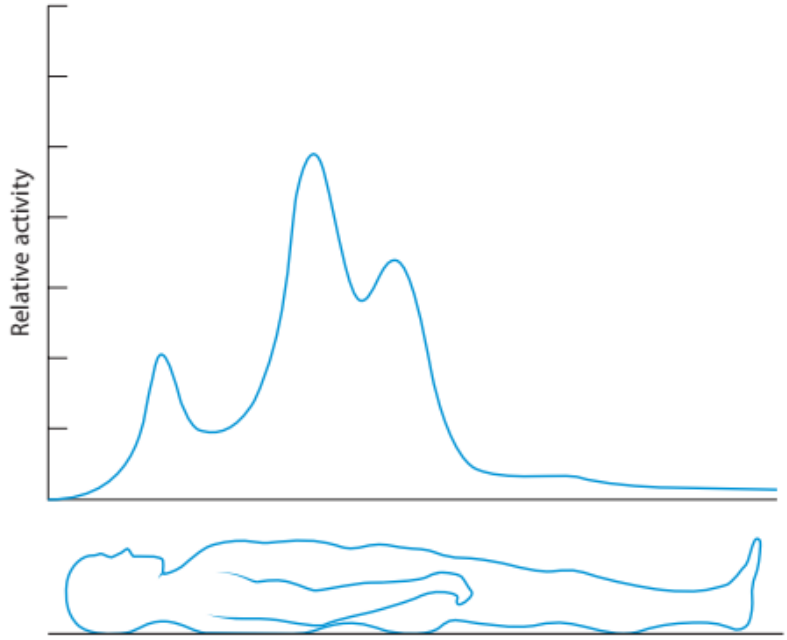

Figure 13-3. Whole-body scan, with a crystal gamma-ray detector, 2 hours after ingestion of Tc. (Reproduced with permission from Beasley TM, Palmer HE, Nelp WB. Distribution and excretion of technetium in humans. *Health Phys.* 1966; 12(10): 1425–1435.)

INDIVIDUAL MONITORING

Personal monitoring for external radiation is the continuous measurement of an individual's exposure dose by means of one or more types of suitable instruments, such as pocket ionization chambers, film badges, electronic dosimeters, and thermoluminescent dosimeters (Chapter 9), which are carried by the individual at all times. The choice of personal monitoring instrument must be compatible with the type and energy of the radiation being measured. For example, a worker who is exposed only to ^{3}H, ^{14}C, or ^{35}S would wear no personal monitoring instrument, since these isotopes emit only beta particles of such low energy that they are not recorded by any of the commercially available personal monitoring devices. In-vitro bioassay procedures would be indicated if personal monitoring were necessary.

Workers who may be exposed to radioactive aerosols may wear a personal air sampler. This usually consists of a cassette that holds a 37-mm-diameter membrane filter through which air is drawn at a rate of 2–5 L per min by a battery-operated pump. The cassette is clipped to the worker's garment near his nose or mouth, and thus produces a breathing-zone sample; the pump, which has an integral air-measuring device, such as a rotameter, is worn on the worker's belt. After exposure, the filter is removed from the cassette, and the activity on its surface is measured in the counting laboratory. Such an open-faced filter collects particles of all sizes. In order to better evaluate the hazard from an aerosol, we often resort to size-selective collectors to collect a sample of "respirable dust." The most commonly used size-selective collector is the cyclone sampler, which is designed to capture particles with $\leq$4-μm AMAD. In this collector, the airstream is forced to travel in a circular path, and particles greater than a given size will be removed from the airstream by centrifugal force. A membrane filter upstream of the cyclone captures the particles that were not removed by the centrifugal force. Since the centrifugal force is a function of the particle speed, the airflow rate of the sampling system must be carefully regulated.

The main purpose of personal monitoring is to obtain information on the exposure of an individual. In addition to this main purpose, personal monitoring is also used to observe trends or changes (in time) in the working habits of a single individual or of a department and thus to measure the effectiveness of a radiation control program. While the distribution of personal-monitoring data might all appear to lie within a normal range when viewed as individual readings, statistical analysis of the grouped data may reveal small but significant differences among different control measures or different operating procedures or work habits that might otherwise have escaped the attention of the radiation safety officer.

RADIATION AND CONTAMINATION SURVEYS

A *survey* is a systematic set of measurements made by a health physicist in order to determine one or more of the following:

1. an unknown radiation source,
2. dose rate,
3. surface contamination, and
4. atmospheric contamination.

In order to make these determinations, the health physics surveyor must choose the appropriate instruments and must use them properly.

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Choosing a Health Physics Instrument

The choice of a surveying instrument for a specific application depends on a number of factors. Some general requirements include portability, mechanical ruggedness, ease of use and reading, ease of servicing, ease of decontamination, and reliability. In addition to these general requirements, health physics survey instruments must be calibrated for the radiation that they are designed to measure, and they must have certain other characteristics.

1. *Ability to respond to the radiation being measured.* This point can be clarified with a practical example: a commonly used side window beta–gamma probe has a window thickness of 30 mg/cm^2. This probe would be worse than useless if one wished to survey for low-energy beta radiation, such as ^{14}C or ^{35}S, or for an alpha contaminant such as ^{210}Po. Each of these radioisotopes emits only radiation whose range is less than 30 mg/cm^2—radiation not sufficiently penetrating to pass through the window of the probe. Incorrect use of this probe, therefore, may falsely indicate safe conditions when, in fact, there may be severe contamination. Similarly, incorrect inferences may be drawn if a neutron monitor is used to measure gamma radiation or if an instrument designed to measure gamma rays is used for neutrons. Another serious source of error may arise if we use particle-counting instrument to measure the radiation level from a pulsed source, such as an accelerator. Consider a machine whose pulse repetition rate is 120 s^{-1}, and the pulse width is 5 microseconds. If we use a particle-counting instrument whose resolving time is greater than the pulse width but less than the time between pulses, the instrument will merely record the pulse repetition rate, 120 cps in this case. It is essential that radiation survey instruments be used only for the radiations they are designed to measure.
2. *Sensitivity.* The instrument must be sufficiently sensitive to measure radiation at the desired level. Thus, an instrument to be used in a search for a lost radium needle should be more sensitive than a survey meter used to measure the radiation levels inside the shielding of an accelerator. In the latter case, where the radiation levels may reach hundreds of milligrays per hour (thousands of milliradians per hour), an ionization chamber whose sensitivity is about 0.01 mGy/h (1 mrad/h) is suitable. In searching for the lost radium needle, however, a sensitivity of 0.01 mGy/h (1 mrad/h) would greatly limit the area that could be covered in the search; a Geiger counter survey meter that has a sensitivity of about 0.5 μGy/h (0.05 mrad/h) is much more useful. For example, if a 1-mg radium needle were lost, the distance within which it could be detected with the ionization chamber is about 90 cm, while the Geiger counter will respond to the lost radium at a distance of 400 cm. In this case, the Geiger counter can thus cover a search area about 20 times greater than the ionization chamber. Too great a sensitivity, however, may be equally undesirable. The range of radiation levels over which the instrument is to be used should be matched by the range of radiation levels for which the instrument is designed. Sensitivity is determined mainly by the value of the input resistor across the detector, *R* in Figures 9-1 and 9-27. The sensitivity of the detector is directly proportional to the size of the input resistance.
3. *Response time*. The response time of a survey instrument may be defined as the time required for the instrument to attain 63% of its final reading in any radiation field. This time is determined by the product of the input capacity (in farads) of the detector and the shunting resistance (in ohms) across the detector, *RC* in Figure 9-1. The time constant is usually expressed in seconds. A survey instrument's time constant may have a strong influence on a radiation measurement. For example, if a measurement made with an

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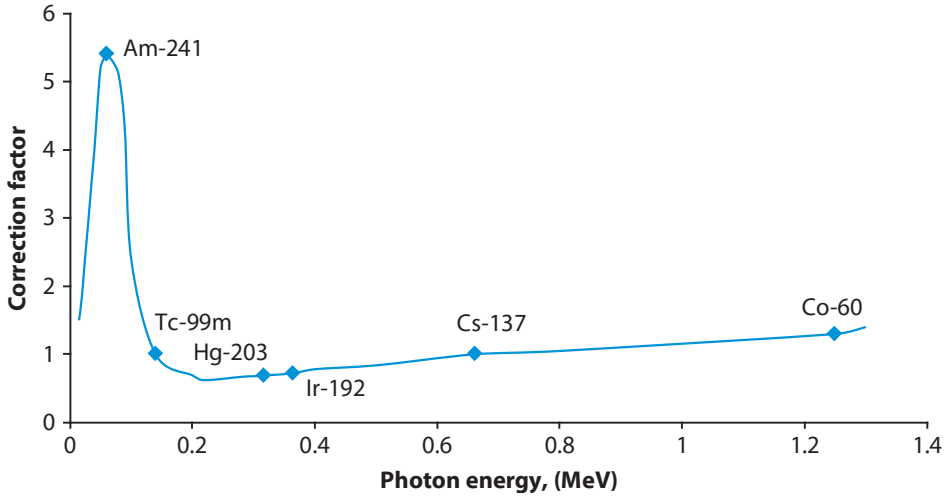

Figure 13-4. Energy dependence of a Geiger counter survey meter. The meter reading is multiplied by the correction factor appropriate to the photon energy in order to obtain the true exposure rate. Figure based on data from Ludlum Model 44-9 GM detector.

ionization-type survey meter whose time constant is 3 seconds is made during a 0.2-second exposure of a diagnostic X-ray is 0.16 mR/h, then the true exposure rate is calculated with an adaptation of Eq. (9.10):

$$\dot{X}(\text{true}) = \frac{\dot{X}(\text{measured})}{1 - e^{-\frac{t}{RC}}} = \frac{0.16 \frac{\text{mR}}{\text{h}}}{1 - e^{-\frac{0.2\,\text{s}}{3\,\text{s}}}} = 2.5 \frac{\text{mR}}{\text{h}}.$$

A low value for the time constant means an instrument that responds to rapid changes in radiation level—such as would be experienced when passing the probe rapidly over a small area of contamination on a bench top or over a small crack in a radiation shield. A fast response time, however, may mean a decrease in sensitivity due to a smaller value of *R*. Furthermore, a fast response time may result in rapid fluctuations of the meter reading, thus making it difficult to obtain an average level. Most instruments offer a range of response times, the appropriate one being selected by the surveyor, who turns the time constant selector switch to the desired value.

4. *Energy dependence*. Most radiation-measuring instruments have a limited span of energy over which the radiation dose is accurately measured. One of the figures of merit of a radiation dosimeter is the energy range over which the instrument is useful. This information must be known by the health physicist in order to choose a proper instrument for a particular application or to interpret the measurements properly. The energy dependence is usually specified by the manufacturer as "accurate to ±10% of the true value from 80 keV to 2 MeV" or by means of an energy-dependence curve (Fig. 13-4). The magnitude of the errors that can arise when the energy dependence factor is overlooked is shown in Table 13-2.

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TABLE 13-2 Energy Dependence of Dose–Rate Response of Geiger–Müller (GM) and Scintillation Counters—Meter Reading for a True Exposure Rate of 1 mR/h

ISOTOPE	GAMMA-RAY ENERGY (MeV)	GM COUNTER (mR/h)	SCINTILLATION COUNTER (mR/h)
^{60}Co	1.25	1.15	0.6
^{226}Ra	0.84	1.0	0.96
^{137}Cs	0.661	0.92	1.39
^{198}Au	0.411	0.82	2.65
^{203}Hg	0.279	1.29	7.5
^{141}Ce	0.145	2.4	14.1
^{241}Am	0.06	6.0	9.8

Source: Reproduced with permission from Peirson DH. Interpretation of instrument readings: Pulse-counting instruments. *Phys Med Biol.* 1963; 7(4):445–453. Copyright © 1963 Institute of Physics Publishing.

Surface Contamination

Surface contamination can be located by scanning with a sensitive detector, such as a thin-end window Geiger counter. Surface contamination is expressed in units of activity per unit area, Contamination limits usually are listed as disintegrations per minute per 100 cm^2 (Table 11-5).

The main hazard from surface contamination is the transmission of the contamination from the surface into the body via inhalation or ingestion. To estimate this hazard, a *smear test* is performed to determine whether the surface contamination is fixed or whether it is loose and therefore transmissible. A smear test consists of wiping the suspected area with a piece of filter paper several centimeters in diameter and then measuring the activity in the paper. The area to be smeared varies according to the extent of the suspected contamination and the physical conditions under which the survey is made; a wipe area of 100 cm^2 is not uncommon. A smear survey is a systematic series of smears used to detect transmissible contamination. It is often done in a work area that is subject to contamination, where the background due to radiation sources is high enough to mask the activity due to contamination, or when detection with a survey meter is difficult, as is the case with ^{3}H, ^{14}C, or ^{35}S. It should be emphasized that a smear test is a qualitative or, at best, a semiquantitative determination whose chief purpose is to allow an estimate to be made of the degree to which surface contamination is fixed. If significant transmissible contamination is found and if, in the opinion of the health physicist, this contamination may be hazardous, then prompt decontamination procedures are instituted.

Decontamination can often be effected simply by wiping the surface with a damp cloth. If this is ineffective, then commercially available chemical solutions may be used. A simple method for removing tactilely transmissible surface contamination is to use masking tape. When the adhesive side of the tape is pressed to the contaminated surface, a good deal of the contaminant will adhere to the tape. Another method, which is more effective than the masking tape is the application of a polymer hydrogel that can seep into tiny crevices to engulf the contaminant. The hydrogel is applied with a roller and after it is dry, the hydrogel skin with the contaminant entrapped in it, is peeled off. Reasonably high decontamination efficiencies, depending on the nature of the surface and the contaminant, can be obtained. The agents that are used for decontamination, of course, become contaminated themselves, and are treated as low-level radioactive waste.

Leak Testing of Sealed Sources

Sealed gamma-ray, beta, bremsstrahlung, and neutron sources are used in a wide variety of applications in medicine, in research laboratories, in industry, and in antiterrorist activities such as examination of baggage and cargo containers. In all cases, the radioactive material is permanently enclosed either in a capsule or another suitable container. Before being shipped from the supplier, all such sources must pass an inspection for freedom from surface contamination and leakage. Either during transport from the supplier or in the course of time, however, the capsule may develop faults through which the radioactive source material may escape into the environment. Because of the potentially serious consequences of such an escape, a sealed source must be tested before being put into use and periodically thereafter for surface contamination and leakage. The testing cycle depends on the nature of the source and on the kind of use to which it is put. However, it is usually recommended that such tests be performed at least once every 6 months. Physically touching the source should never be attempted, as the dose rates on contact with sources can be exceptionally high. The following techniques may be employed to perform these tests:

1. Wipe the source with either a piece of wet filter paper or a cotton swab. Repeat at least 7 days later. If less than 200 Bq (0.005 μCi) alpha or less than 2000 Bq (0.05 μCi) beta activity are wiped off each time, the source is considered free of leaks.
2. For high-activity sources such as those used in teletherapy, where wiping the source might be hazardous, accessible surfaces of the housing port or collimator should be wiped while the source is in the "off" position.
3. Immerse the source in ethylene glycol (ethanediol) and reduce the pressure on the liquid to 100 mm Hg for a period of 30 seconds. A leak is indicated if a stream of fine bubbles issues from the source. This method is reliable only for such sources where enough gas would be trapped to produce a stream of fine bubbles.