1
Introduction

Health physics or radiation protection is the science dealing with the protection of radiation workers and the general public from the harmful effects of radiation. Health physicists work in a variety of environments, including medical facilities, facilities utilizing nonionizing radiation, universities, accelerator complexes, power reactors, and fuel cycle facilities. The health physicist is responsible for the radiological safety aspects of facility equipment and services. Radiological assessments of plant equipment, facility modifications, design changes, employee exposures, or the assessment of radiological effluents are key functions of a health physicist.

The fundamental tools of the health physicist include the fields of mechanics, electricity and magnetism, energy transfer, quantum mechanics. Atomic and nuclear structure, radioactive transformations, and the interaction of radiation with matter are the cornerstones of health physics knowledge. Application of these fundamental tools permits the health physicist to measure, quantify, and control radiation exposures to affected groups.

Introductory health physics texts typically cover these topics in several hundred pages. Because the scope of this text builds upon these fundamental concepts, we will not repeat them herein. The reader is referred to the texts listed as references to this chapter for a discussion of health physics fundamentals. We will, however, provide several appendices that illustrate selected fundamental concepts. Also included is an extensive set of scenarios, including over 160 worked examples, that illustrate the fundamental concepts and permit the reader to assess his or her knowledge of these concepts. Because the fundamentals are needed to fully understand the remaining chapters in this text, a review of the scenarios in this chapter is recommended.

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1.1 Scenarios

Scenario 1.1

One of your neighbors, while digging up his back yard to build a pool, has discovered some old planks. Another neighbor, who has been investigating the possibility of the existence of a Viking settlement in the area, believes that the planks may be significant. He wishes to conduct an archeological expedition prior to any further construction. You offer to carbon date the wood to help settle the argument.

1.1 Carbon dating is possible because:
   a. The specific activity of carbon-14 in living organisms has changed over time, and one can identify the era of time the organism lived based on its current specific activity.
   b. Carbon-14 is in secular equilibrium with its daughter.
   c. The specific activity of carbon-14 in living organisms is relatively constant through time, but decays after the death of the organism.
   d. The specific activity of carbon-14 in wood increases over time due to shrinkage of the wood.

1.2 Calculate the approximate age of the wood given the following:

C-14 $T_{1/2}$ = 5715 years

Specific activity for C-14 in a nearby living tree = 1.67×10^{-1} Bq/g

Specific activity for C-14 in the old wooden plank = 1.50×10^{-1} Bq/g

Scenario 1.2

A nearby hospital has received a shipment of a Mo-99 generator. The shipment contained 1000 mCi of Mo-99 when manufactured. It arrived at the hospital 48 h after its production. The decay scheme is illustrated in Figure 1.1.

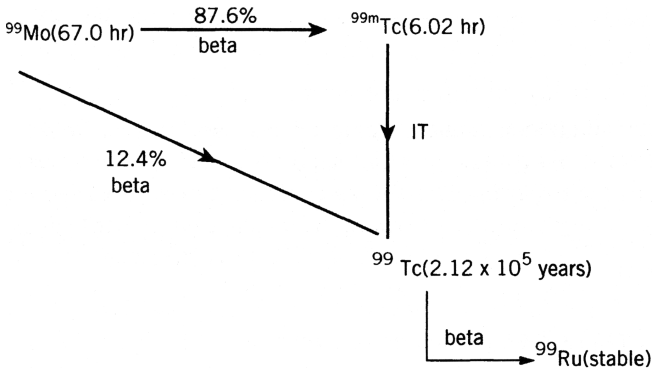

Figure 1.1 Decay scheme for Mo-99.

1.3 If the generator is milked exactly upon arrival at the hospital, how much Tc-99m will be obtained? Assume that 95% of the available Tc-99m is eluted.

1.4 If the generator is milked 24 hr after the initial milking, how much Tc-99m will be obtained?

Scenario 1.3

Consider a parent radioisotope A ($T_{1/2}$ = 10 hr) that decays to a daughter radioisotope B ($T_{1/2}$ = 1 hr).

1.5 Which of the following statements is true concerning these radioisotopes?

a. Because $\lambda_A > \lambda_B$, the parent and daughter will eventually reach the condition of transient equilibrium.
b. Because $\lambda_A >> \lambda_B$, the parent and daughter will eventually reach the condition of secular equilibrium.
c. Because $\lambda_A = \lambda_B$, no state of equilibrium can ever exist between the parent and daughter.
d. Because $\lambda_B > \lambda_A$, the parent and daughter will eventually reach the condition of transient equilibrium.
e. Because $\lambda_B >> \lambda_A$, the parent and daughter will eventually reach the condition of secular equilibrium.

1.6 Assuming that the activity of the daughter is zero at time zero, at what time (t) will the daughter reach its maximum activity?

Scenario 1.4

The plant manager at your facility has requested that you review the following questions and provide the best solution. These questions will be used to assess the qualification of health physics candidates for entry-level positions in your facility's radiological controls department.

1.7 Tissue dose from thermal neutrons arises principally as a result of:

a. (n, γ) reactions with hydrogen.
b. (n, γ) reactions with hydrogen and (n, p) reactions with nitrogen.
c. (n, p) reactions with carbon.
d. (n, α) reactions with carbon.
e. (n, α) reactions with carbon and (n, γ) reactions with hydrogen.

1.8 Tissue dose from fast neutrons (0.1 to 14 MeV) is due principally to:

a. Resonance scattering with nuclei.
b. Inelastic scattering with nuclei.
c. Coulomb scattering with nuclei.
d. Nuclear capture and spallation.
e. Elastic scattering with nuclei.

1.9 The most probable process for energy deposition by a 1-MeV photon in tissue is:
   a. Photoelectric absorption.
   b. Pair production.
   c. Compton scattering.
   d. Photonuclear absorption.
   e. Bremsstrahlung.

1.10 The principal mechanism of dose deposition by a 5-MeV alpha particle that stops in tissue is:
   a. Inelastic scattering by atomic electrons.
   b. Elastic scattering by atomic electrons.
   c. Elastic scattering by atomic nuclei.
   d. Inelastic scattering by atomic nuclei.
   e. Nuclear spallation.

1.11 The principal mechanism of dose deposition by a 100-keV beta particle that stops in tissue is:
   a. Elastic scattering by atomic electrons.
   b. Elastic scattering by atomic nuclei.
   c. Inelastic scattering by atomic nuclei.
   d. Inelastic scattering by atomic electrons.
   e. Bremsstrahlung.

1.12 The average number of ion pairs produced by 100-keV beta particle that stops in air is approximately:
   a. 300
   b. 30
   c. 30 000
   d. 3000
   e. 300 000.

1.13 The average number of ion pairs produced by a 100-keV beta particle that stops in a germanium semiconductor is:
   a. 30 000
   b. 30
   c. 300
   d. 3000
   e. 300 000.

1.14 A nuclide that undergoes orbital electron capture:
   a. Emits an electron, a neutrino, and the characteristic X-rays of the daughter.
   b. Emits a neutrino and the characteristic X-rays of the daughter.
   c. Also decays by positron emission.
   d. Also emits internal conversion electrons.
   e. Makes an isomeric transition.

1.15 The specific gamma-ray emission rate for Cs-137 in units of R hr^{-1} Ci^{-1} m^2 is approximately:
   a. 1.3

b. 0.12
c. 0.33
d. 0.05
e. 0.77.

1.16 An example of an organ or tissue for which the Annual Limit on Intake (ALI) is determined by the limit for nonstochastic effects is the:
a. Red bone marrow.
b. Gonads.
c. Lung.
d. Breast.
e. Thyroid.

Scenario 1.5

The radioisotope I-126 (atomic number 53) can decay into stable Te-126 (atomic number 52) by orbital electron capture (EC) or by positron emission. It can, alternatively, decay by negative beta emission into stable Xe-126 (atomic number 54). The fractions of the transformations that take place via these modes are: EC 55%, positron emission 1%, and beta decay 44%. An I-126 source also emits gamma photons of energy 386 keV and 667 keV as well as characteristic X-rays of Te. The energy equivalents (Δ) of the mass excesses of the atoms involved in these transformations are (Δ = atomic mass – atomic mass number):

Atom	Δ (MeV)
Te-126	–90.05
I-126	–87.90
Xe-126	–89.15

The energy equivalent of the electron rest mass is 0.511 MeV, and the binding energy of the K-shell electron in I-126 is 32 keV.

For the following questions, choose the best answer.

1.17 The energy release (Q-value) by the decay of I-126 via capture of a K-shell electron, going directly to the ground state of Te-126, is:
a. 0.03 MeV.
b. 1.13 MeV.
c. 2.12 MeV.
d. 2.15 MeV.
e. 2.18 MeV.

1.18 The energy released (Q-value) by the decay of I-126 via position emission to the ground state of Te-126 is:
a. 0.51 MeV.
b. 1.02 MeV.

c. 1.13 MeV.
d. 1.64 MeV.
e. 2.15 MeV.

1.19 The energy released in the decay of I-126 to the ground state of Xe-126 by beta emission is:
a. 0.20 MeV.
b. 0.23 MeV.
c. 0.90 MeV.
d. 0.74 MeV.
e. 1.25 MeV.

1.20 Of the following kinds of radiation emitted from I-126, which is the single least significant potential contributor to internal dose?
a. Annihilation photons.
b. Bremsstrahlung.
c. Internal-conversion electrons.
d. Auger electrons.
e. Antineutrino.
How would your answer change if external dose contributions were under consideration?

1.21 Why are the 32-keV Te X-rays present with an I-126 source?
a. The nucleus of Te-126 has excess energy after the EC event. This excess energy is released by Te-126 as X-rays.
b. Stable Te-126 has excess energy after the positron emission. This excess energy is released by Te-126 as X-rays.
c. Electrons rearranging between the L and M shells produce X-rays.
d. Te X-rays are released when the EC event creates a vacancy in the inner shells, and electrons from outer shells fill the vacancy.
e. Te X-rays are equivalent to the bremsstrahlung radiation emitted by I-126.

Scenario 1.6

The nuclide Sr-90 (atomic number 38) decays by beta emission into Y-90 (atomic number 39), which then decays by beta emission into Zr-90 (atomic number 40), with the half-lives noted below:

$$\text{Sr-90} \xrightarrow[27.7\text{ years}]{} \text{Y-90} \xrightarrow[64.2\text{ hr}]{} \text{Zr-90}$$

1.22 What is the mean, or average, lifetime of a Y-90 atom?
a. 31.1 hr.
b. 44.5 hr.
c. 77.04 hr.
d. 92.6 hr.
e. 128.4 hr.

1.23 What is the specific activity of Y-90 in SI units?
   a. 5.42×10^5 Bq/kg
   b. 7.22×10^{16} Bq/kg
   c. 2.01×10^{19} Bq/kg
   d. 7.22×10^{19} Bq/kg
   e. 6.49×10^{21} Bq/kg.

1.24 Starting with a pure Sr-90 sample at time $t = 0$, a researcher finds that the Y-90 activity is 3.4 MBq at $t = 72.0$ hours. What was the activity of the Sr-90 at $t = 0$?
   a. 1.84 MBq
   b. 3.40 MBq
   c. 4.37 MBq
   d. 6.29 MBq
   e. 7.39 MBq.

Scenario 1.7

You have been asked to assist in the technical evaluation of an ionization chamber and environmental sampling results. Your boss has requested answers to the following questions. Assume the density of air at STP = 1.293×10^{-6} kg cm^{-3}.

1.25 A free air ionization chamber shows a flow of electrical charge of 1×10^{-9} A. The chamber has a sensitive volume of 4 cm^3. The reading is taken at 10 °C and 755 mm Hg. Find the exposure rate in R/s based on STP conditions.

1.26 You are asked to provide immediate, on-site measurement results for a series of environmental samples that are being collected every 100 min. It has been requested that each sample count be preceded by a background count. From past experience, you estimate that the net sample and background counting rates should be approximately 2400 and 300 cpm, respectively. Assuming that each sample must be analyzed before the next one is received, how long would you count the sample to minimize the standard deviation estimate for the sample's net activity?

1.27 A water sample that was counted for 10 min yielded 600 counts. A 40-min background count yielded a background rate of 56 cpm. At a 95% confidence level (one-tail test), determine whether or not there was any net activity in the sample.

Scenario 1.8

You are responsible for operating the counting room at a nuclear facility. You need to minimize the counting time required for air samples because of the heavy workload and a need to streamline operations in the count room. The bulk of your air sample workload is counting I-131. The following parameters are applicable to your operation:

Counting efficiency = 20%

Background count time = sample count time

Background count rate = 50 cpm

Sampling flow rate = 5 liters/min

Sample collection time = 10 min

Iodine collection efficiency = 70%

DAC for iodine = 900 Bq/m^3

1.28 Calculate the minimum sample and background counting time required to ensure an LLD at the 95% confidence level less than or equal to 0.10 DAC for I-131.

1.29 List methods that could be used in the field or in the counting room to reduce the time required to process I-131 samples. Explain how each method reduces processing time.

Scenario 1.9

As a health physicist at a nuclear facility, you are asked to develop a program to characterize the radioactive particulate emissions through the facility's main ventilation stack. The following questions relate to various aspects of this assignment.

1.30 In designing the sampling system, you have determined that the stack internal diameter is 0.5 m and the volumetric flow through the stack is 20 m^3 min^{-1}. You want to use a vacuum source which will provide a constant volumetric flow of 200 liters/min through your sampling train. Assuming laminar flow, what should the internal diameter of the sampling nozzle be to ensure isokinetic sampling conditions?

1.31 To ensure that your sample is representative of laminar flow conditions (nonturbulent, constant velocity) within the stack, discuss factors that you should consider relative to the location of your sampling nozzle within the stack.

1.32 You have decided to use filtration techniques to capture your sample and are evaluating three types of media (cellulose, glass-fiber, and membrane filters). List advantages and disadvantages of each.

Scenario 1.10

You are responsible for a high-volume air sampler located downwind from a Department of Energy (DOE) facility following a suspected release of Pu-239. The air sampler has a calibrated volumetric flow rate of 55 SCFM, and the filter has an alpha self-absorption factor and filter collection efficiency of 0.4 and 0.8, respectively. The air sampler is operated at this flow rate for 1 hr, and the filter surface is measured with a gross alpha probe detector having an active detection

area of 60 cm^2 and a background count of 20 counts in 100 min. The detector efficiency for alpha is 0.3 cpm dpm^{-1}, and the active filter area is 500 cm^2. Assume that the filter face velocity is uniform.

Data

Half-Life for Pu-239 = 24 100 years

Alpha yield = 100%

LLD (95%) = 4.66 σ_b (where σ_b is the standard deviation of the background)

1.33 The initial filter-face alpha count immediately after the 1 hr sampling period was 2000 for a 10-min counting interval. Forty-eight hours later, the same filter is measured again with the same detector, and the count was 220 in 100 min. Explain why the count rate is lower 48 hr later.

1.34 What is Pu-239 airborne activity (in dpm/m^3) and the standard deviation for this measured quantity?

1.35 What is the lower limit of detection (LLD) at the 95% confidence level for this air sampling and detection system (in dpm/m^3) for the same sampling conditions?

Scenario 1.11

This scenario deals with the working-level unit.

With the passage of the Radon Control Act of 1988, the Environmental Protection Agency (EPA) is now instructed by the Congress to assess public risks of radon exposure in public buildings (including schools) throughout the nation. Regarding the measurement, detection, and health physics of radon-222 and its daughter products, answer the following questions:

1.36 Historically, an operational definition of the working-level exposure unit (WL) for radon-222 daughters has been 100 pCi/liter of each short-lived daughter product in secular equilibrium. Using this definition and the data provided derive the total alpha energy per liter of air (MeV/liter) associated with a concentration of one working level. Radon and its short-lived daughters include:

Nuclide	Alpha Energy (MeV)	Half-life
Radon-222	5.49	3.82 days
Polonium-218	6.00	3.05 min
Lead-214	0	26.8 min
Bismuth-214	0	19.7 min
Polonium-214	7.68	1×10^{-6} min

1.37 Using the data provided, calculate the concentration of radon-222 gas in air determined from a single-count, filter collection method for radon daughters. Assume a 50% equilibrium between radon-222 and its daughters. Neglect special considerations for radioactive growth and decay during sampling and counting. The following data are provided:

Sample collection period = 5 min

Counting time = 1 min

Total alpha counts = 230

Counting efficiency = 0.3

Pump flow rate = 10 liters min^{-1}

Conversion factor = 150 dpm alpha $liter^{-1}$ WL^{-1}

1.38 List common methods for the detection and measurement of radon and/or its daughters for use in assessing public exposure in building structures.

Scenario 1.12

A common type of portable beta–gamma survey instrument uses an air ionization chamber vented to atmospheric pressure. The cylindrical detector is 3 in. high and 3 in. in diameter with a 7-mg/cm^2 beta window and a 400-mg/cm^2 beta shield. The side walls are 600 mg/cm^2. Answer the following questions with respect to the instrument's response versus the 'true' dose rates specifically associated with the following conditions.

1.39 Briefly describe a potential source of error associated with measuring gamma and beta dose rates while moving in and out of a noble gas environment.

1.40 List and briefly explain two harsh environmental conditions which could have an adverse effect on the accuracy of the instrument response while in the area.

1.41 Briefly describe the most significant source of error associated with measuring true beta and gamma surface dose rates from contact measurements of small sources.

1.42 Briefly explain a source of error associated with measuring beta dose rates from large-area sources, with each source comprised of a different radionuclide.

1.43 Briefly describe a source of error associated with measuring beta dose rates from high-energy beta sources using open minus closed window readings.

Scenario 1.13

ANSI N13.11-1983, 'American National Standard for Dosimetry – Personal Dosimetry Performance Criteria for Testing', is used as a basis for testing the performance of suppliers of dosimetry services. This standard provides criteria for testing personnel dosimetry performance for any type of dosimeter whose reading is used to provide a lifetime cumulative personal radiation record. The test procedure in this standard evaluates the absorbed dose and dose equivalent at two irradiation depths (0.007 cm and 1.0 cm). The radiation sources used for the performance tests are Cs-137, Sr-90/Y-90, heavy water moderated Cf-252, and an X-ray machine. The X-ray machine is used to generate several photon beams with average energies between 20 keV and 70 keV. Choose the single answer which is most correct.

1.44 The provisions of this standard apply:
   a. to neither pocket dosimeters nor extremity dosimeters.
   b. to pocket dosimeters but not to extremity dosimeters.
   c. only to beta and gamma radiation.
   d. to extremity dosimeters but not to pocket dosimeters.
   e. to film badges but not to thermoluminescent dosimeters (TLDs).

1.45 Because of the particular irradiation depths chosen for the tests, a dosimetry system which is calibrated with the standard tests may be reporting doses which are different than the actual dose received. For which of the following tissues (red bone marrow, skin, gonads, lens of the eye, or whole body) is this difference most significant?

1.46 Because of the particular radiation sources specified, the standard least adequately tests for radiations emitted by:
   a. C-14, power reactor leakage neutrons.
   b. P-32, Cf-252.
   c. Y-90/Sr-90, Am–Be source.
   d. Co-60, Ni-65.
   e. Uranium slab, Cf-252.

1.47 A dosimeter of a processor who has passed the test category for:
   a. beta radiation, is appropriate for measuring low-energy photons.
   b. beta radiation, is not appropriate for measuring beta radiation from all sources.
   c. low-energy photons, can be used to pass the performance test for beta radiation.
   d. high-energy photons and the category for low-energy photons, can be assumed to pass the test for mixtures of high-energy and low-energy photons.
   e. neutrons, is appropriate for measuring neutron radiation from any sources.

1.48 This standard:
   a. forms the basis for the National Voluntary Laboratory Accreditation Program for dosimetry processors.
   b. provides guidance for individual variability from reference man.
   c. provides guidance for summing the internal and external dose.
   d. is applicable to the entire range of gamma energies.
   e. is not required to be implemented by 10 CFR 20.

Scenario 1.14

For each of the situations below (1.49 to 1.53), select the personnel dosimeter which is most suitable for the purpose of establishing primary dose records. In each case substantiate your choice of dosimeter. Limit your choice of dosimeter to the following:

1. A common film badge with 300 mg/cm^2 plastic filtration over all areas except for the 14-mg/m^2 mylar window.
2. A TLD albedo containing both Li-6 and Li-7 elements.
3. A TLD albedo containing both Li-6 and B-11 elements.
4. A calcium sulfate, manganese-activated TLD element in a tissue equivalent holder.
5. A proton recoil film badge.
6. A four-element TLD with lithium borate phosphors, 300-mg/cm^2 plastic filtration over two elements, aluminum over the third element, and lead over the fourth element.
7. A four-element TLD with lithium borate phosphors, a thin mylar filter over one element, plastic filters over two elements, and an aluminum filter over the fourth element.
8. A natural LiF TLD element.
9. A calcium sulfate, dysprosium-activated TLD element in a tissue equivalent holder.
10. A two-element TLD with lithium borate phosphors and 300-mg/cm^2 plastic filters.

1.49 An accelerator facility using tritiated targets with 14-MeV deuteron beams.
1.50 A mixed neutron and gamma field where gamma dose predominates.
1.51 A radiographer using a 320-kVp X-ray machine.
1.52 A field of high-energy, 6-MeV photons.
1.53 A field of mixed beta (average energy of 200 keV) and gamma (average energy of 800 keV) radiation.

Scenario 1.15

You supervise an in-house TLD system for occupationally exposed workers governed by US Nuclear Regulatory Commission regulations. The TLD badge consists of two LiF chips of 235-mg/cm^2 thickness. Chip 1 is covered by 7 mg/cm^2 of plastic, and Chip 2 is shielded by 850 mg/cm^2 of lead and 150 mg/cm^2 of plastic. The TLD system is calibrated by exposing badges to known quantities of beta and gamma radiations and plotting the TL reader output versus mrem dose equivalent. Both the gamma and beta calibration curves are linear and pass through the origin (0, 0) on the graph. The gamma calibration curve indicates that 6000 TL units equals 500 mrem of gamma dose equivalent, and the beta curve yields 750 TL units per 1000 mrem of beta dose equivalent.

The following data are provided:

1. The control dosimeter reads 120 TL units on both Chips 1 and 2. (Both Chips have the same gamma sensitivity.)
2. Chip 1 = 12 270 TL units.
   Chip 2 = 11 520 TL units.
3. The beta calibration curve for other tissue depths includes the following:

Tissue Depth (mg/cm^2)	Percentage of Dose Equivalent at 7 mg/cm^2
7	100
100	50
300	25
500	10
1000	1

4. The gamma dose equivalent remains constant at tissue depths from 7 to 1000 mg/cm^2.

1.54 Calculate the skin and whole-body dose equivalents for the exposed TLDs noted above in item 2.
1.55 Calculate the dose to the lens of the eye.
1.56 Explain if any dose limits were exceeded. Justify your answer by stating the limits and identifying the source of the limits that you applied.

Scenario 1.16

A facility is in the process of setting up a neutron dosimetry program. You have been asked to consult on this matter. The following dosimeters are under consideration: TLD, recoil track-etch, neutron track type A (NTA) film, and bubble detectors. A final option is to use stay time calculations based on survey results from a 'rem-ball' that has been calibrated using D_2O-moderated Cf-252.

1.57 Which one of the following statements is incorrect?
   a. There is no neutron dosimetry system in use today that is adequate (±50% of the true dose equivalent) for all situations where neutron dosimetry is required.
   b. The neutron quality factor between 0 and 20 MeV is relatively constant at a value of about 10.
   c. Neutron energies can span nine decades in some monitoring situations.
   d. Neutron monitoring is usually performed in a mixed field of neutron and gamma radiation.
   e. Stay-time calculations, though often used, may be unreliable due to variations of neutron dose rates and energies in a given neutron radiation area.

1.58 In a field of mixed neutron and gamma radiation, the gamma dose measured on a phantom is:
   a. greater than the gamma dose measured in air due to the $H(n, \gamma)\, D$ reaction in the phantom.
   b. less than the dose measured in air due to the moderation of neutrons in the phantom.
   c. the same as the measured dose in air because phantoms do not influence gamma irradiation.
   d. less than the dose measured in air because some incident gamma rays are absorbed in the phantom.
   e. not a quantity of interest in a dosimetry program.

1.59 If no corrections are made to the dosimeter response for neutron energy, TLD albedo dosimeters calibrated with a bare Cf-252 source will:
   a. give accurate indications (±50%) of neutron dose equivalent in soft (thermal or epithermal) spectra.
   b. underestimate the neutron dose equivalent by as much as a factor of 2 in soft (thermal or epithermal) spectra.
   c. overestimate the neutron dose equivalent regardless of the incident spectrum.
   d. underestimate the neutron dose equivalent regardless of the incident spectrum.
   e. overestimate the neutron dose equivalent in soft (thermal or epithermal) spectra.

1.60 Which one of the following statements is true regarding neutron bubble detectors?
   a. They are insensitive to intermediate-energy neutrons.
   b. They are accurate within ±30% in neutron dose rates of over 1000 rad/hr.
   c. They are affected by temperature.
   d. They cannot measure the total integrated dose.
   e. They are not yet commercially available.

1.61 Which of the following choices would most accurately measure the neutron dose equivalent for commercial power reactor containment entries?
   a. A TLD albedo dosimetry system calibrated to D_2O-moderated Cf-252.
   b. A TLD albedo dosimetry system calibrated to AmBe.
   c. A proton-recoil dosimetry system calibrated to D_2O-moderated Cf-252.
   d. A proton-recoil dosimetry system calibrated to AmBe.
   e. An NTA film dosimetry system calibrated to D_2O-moderated Cf-252.

Scenario 1.17

A large community hospital wishes you to set up a personnel monitoring program. The following organization information is provided:

Department A: The nuclear medicine department is a well-equipped department using technetium-99m for all its studies. The Tc-99m is milked from a generator, and the radiopharmaceuticals are prepared within the nuclear medicine department. The department has sealed sources of cobalt-57, cesium-137, and barium-133 for calibrating the dose calibrator.

Department B: The X-ray department is an active group using fluoroscopic procedures, general diagnostic X-ray procedures, and some special procedures.

Department C: The radiation therapy department is an active group using a Co-60 teletherapy device and a 4.0-MeV linear accelerator, but no brachytherapy.

Department D: The research department is a fairly active department using only hydrogen-3 and carbon-14.

1.62 What departments will require personnel monitoring for photons?
1.63 What department will require personnel monitoring for neutrons?
1.64 What departments will benefit from both a personnel monitor at the belt (under leaded apron) and one at the collar?
1.65 What department would need ring badges?
1.66 In what department would the assessment of skin dose be important?
1.67 What department might require bioassay?
1.68 List positive characteristics of film dosimeters for personnel monitoring.
1.69 List negative characteristics of film dosimeters for personnel monitoring.
1.70 List positive characteristics of TLDs for personnel monitoring.
1.71 List negative characteristics of TLDs for personnel monitoring.

Scenario 1.18

This scenario involves the properties of gas-filled detectors.

Data

Air density = 1.29 kg m^{-3} at STP

1 torr = 1 mm Hg at 0 °C

1.72 Consider two cylindrical gas ionization chambers, A and B. The chamber of detector A has the dimensions 0.5 cm in diameter and 5 cm in height. Detector B has the dimensions 1.0 cm in diameter and 5 cm in height. both detectors have the same chamber wall material and thickness, fill gas, and chamber pressure. If detector A shows an output current of 1.0×10^{-10} A when placed in an isotropic gamma field, what theoretical response should be given by detector B when placed in the same field? Neglect detector end effects.
   a. 2.5×10^{-11} A
   b. 4.0×10^{-10} A
   c. 2.0×10^{-10} A
   d. 5.0×10^{-11} A
   e. 1.0×10^{-10} A.

1.73 The gas fill pressure in detector A is 7600 torr, and the detector sensitivity is 1.2×10^{-10} A-hr/R. What would the detector sensitivity be if the gas fill pressure was increased to 11 400 torr?

1.74 Assuming a chamber pressure of 7600 torr, a chamber volume of 100 cm^3, and a temperature of 20 °C, calculate the dose equivalent rate in mSv/hr for a tissue equivalent wall ion chamber if the saturated ion current is 9.0×10^{-14} A. For this question, assume 100 mR = 100 mrem = 1 mSv.

1.75 An ambient-pressure air ion chamber is calibrated at 7000-feet altitude in New Mexico at 20 °C, 591.6-torr air pressure, to read correctly under those conditions. What dose equivalent rate will it indicate in a 1 mSv/hr field at sea level in the Marshall Islands at 36 °C, 760.0-torr air pressure?
   a. 0.74 mSv/hr
   b. 0.82 mSv/hr
   c. 1.00 mSv/hr
   d. 1.22 mSv/hr
   e. 1.36 mSv/hr.

Scenario 1.19

You are the station health physicist at a nuclear power station. The Chemistry Manager has asked you to review a purchase requisition for an N-16 calibration source. The source generates N-16 via an (α, p) reaction involving 160 mCi of curium-244 and carbon-13. The source gamma emission strength is 2.2×10^6 gammas/s, and the neutron emission strength is 2.0×10^5 neutrons/s. Assume a gamma energy of 6.1 MeV and an average neutron energy of 2.5 MeV. The following information is provided for your evaluation:

Physical Quantity (6.1 MeV)	Water	Air	Muscle	Lead
Density (g/cm^3)	1.00	0.001293	1.0400	11.35
Mass-energy absorption coefficient (cm^2/g)	0.0180	0.0163	0.0178	—
Mass-attentuation coefficient (cm^2/g)	0.0277	0.0252	0.0274	0.0435

Point Source Dose Buildup Factors in Lead (*ux*).

Energy = 6.0 MeV	1	2	4	7	10	15	20
	1.18	1.40	1.97	3.34	5.69	13.8	32.7

The neutron flux to dose equivalent (k) at 2.5 MeV is $k = 20$ n/cm^2-s = 2.5 mrem/hr.

1.76 Calculate the total gamma dose equivalent rate at 1 ft. Assume a 100% emission rate from the principal gamma peak.
1.77 Calculate the total neutron dose equivalent rate at 1 ft.
1.78 Lead and polyethylene are available to shield the source. How would you arrange these materials to yield the lowest overall dose rate?
   a. Polyethylene followed by lead.
   b. Lead only.
   c. Polyethylene only.
   d. Lead followed by polyethylene.
   e. No shielding is necessary because the 12-in. air gap will sufficiently scatter/attenuate the neutrons.
1.79 What is the shielding requirement (cm of lead) to reduce the gamma dose rate at 1 ft by a factor of 5?

Scenario 1.20

As the HP supervisor at a reactor decommissioning project, the project engineer has asked you to assist in the evaluation of methods to reduce radiation levels emanating from a neutron-activated concrete shield to meet release limits for unrestricted use. The preferred method requires you to predict the depth to which a slab of neutron-activated concrete must be excavated to allow free release. Other methods she has asked to be evaluated include delayed decommissioning and the addition of shielding.

Assume that the neutron relaxation length in concrete is 15 cm. The current exposure rate is 20 μR/hr 1 m from the slab. The only applicable release limit is 5 μR/hr 1 m from the surface. The concrete source term, based upon a single concrete core sample 1 in. deep, is as follows:

Activation Data for Concrete Source Term.

Nuclide	Radiations			Specific Activity (pCi/g)	Gamma Constant (R/Ci-hr @ 1 m)
	Decay Mode	Energy (MeV)	$T_{1/2}$		
H-3	Beta	0.0186	12.3 years	1000	—
C-14	Beta	0.156	5715 years	500	—
Mn-54	Gamma	0.835	312 days	2500	0.47
Co-60	Gamma Gamma Beta	1.332 1.173 0.314	5.27 years	2500	1.32

1.80 For each of the following three methods for meeting the release limit, list two advantages and two disadvantages:
1. Time to allow decay.
2. Immediate removal.
3. Add shielding.

1.81 Based upon the data provided, estimate the depth of the excavation required to allow free release.

1.82 Assuming that no concrete removal occurs, predict the time necessary to allow the principal radionuclide of interest, Co-60, to decay to the release limit. Assume that the 20 μR/hr exposure rate is due solely to the Co-60.

1.83 How much more concrete shielding would be needed to reduce the exposure rate at 1 m to the release limit? Neglect the geometry considerations. The mass attenuation coefficient is 0.06 cm^2/g, and the density of concrete is 2.5 g/cm^3. Buildup is assumed to be a constant factor of 2.

Scenario 1.21

You are involved in an assessment of the results of an activation experiment that produced Na-24. The buildup and decay of this source and the resultant dose rates require your attention. Answer the following questions regarding the shielding and activation of the Na-24 source.

1.84 What is the flux in particles per square centimeter per second which will produce 4.0×10^7 Bq of Na-24 at saturation in an aluminum target of 1-cm^2 cross-section and 1-g weight? Assume that the production cross-section is 20 mb.

Data

Atomic weight of aluminum = 27

Avogadro's number = 6.02×10^{23}

1 barn = 1.0×10^{-24} cm^2

1.85 Immediately after an irradiation time of 30 hr, what would be the amount of Na-24 present? Assume no initial activity. The half-life of Na-24 is 15 hr.

1.86 What is the dose equivalent rate to a person standing 1 m from the unshielded Na-24 source in air? The following information should be considered in your answer:

Gamma 1 = 1.4 MeV @ 100%

Gamma 2 = 2.8 MeV @ 100%

Air density = 0.00129 g/cm^3

Energy absorption coefficient = 2.3×10^{-5} cm^{-1} @ 1.4 and 2.8 MeV

Assume 1 mrem = 1 mrad

1 MeV = 1.6×10^{-6} erg

Scenario 1.22

You are a consulting health physicist. A client plans to build a 10-MCi Co-60 irradiation facility to sterilize surgical equipment. You may assume that the activity is in the form of a point source. The following data may be useful:

Co-60 gamma constant = 13.2 R-cm^2 hr^{-1} mCi^{-1}

Table of Linear Attenuation Coefficients and Fluence Buildup Factors for Concrete.

Energy (MeV)	u (cm^{-1})	B			
		$ux = 7$	$ux = 10$	$ux = 15$	$ux = 20$
0.5	0.204	16.6	29.0	58.1	98.3
1.0	0.149	11.7	18.7	33.1	50.6
1.17	0.140	11.0	17.5	30.6	46.4
1.25	0.135	10.7	16.9	29.4	44.4
1.33	0.130	10.4	16.3	28.2	42.4
1.5	0.121	9.7	15.0	25.7	38.2

1.87 What would be the exposure rate at a distance of 3 m from the unshielded 10-MCi Co-60 source?

1.88 When exposed, the source will be in the center of a room having internal dimensions of 5 m × 5 m × 5 m with walls of 1-m-thick concrete. The room layout is illustrated in Figure 1.2. Based upon this information, calculate the maximum photon fluence rate in photons cm^{-2} s^{-1} at a point on the exterior surface of the shield wall. State any assumptions used. Ignore scatter off air or walls other than the wall between the source and the reference point.

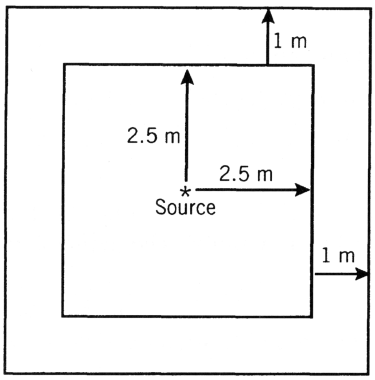

Figure 1.2 Proposed irradiation facility floor plan.

1.89 The buildup factor should only be used:
- a. for photons of energy below 3 MeV.
- b. for photons of energy above 0.5 MeV.
- c. in cases where the shield thickness exceeds 3 relaxation lengths.
- d. for situations involving 'broad-beam' or 'poor' geometry.
- e. for situations involving 'narrow-beam' or 'good' geometry.

Scenario 1.23

Consider both broad and narrow beams of 1-MeV photons, illustrated in Figure 1.3, that are normally incident on different thicknesses of uranium slabs.

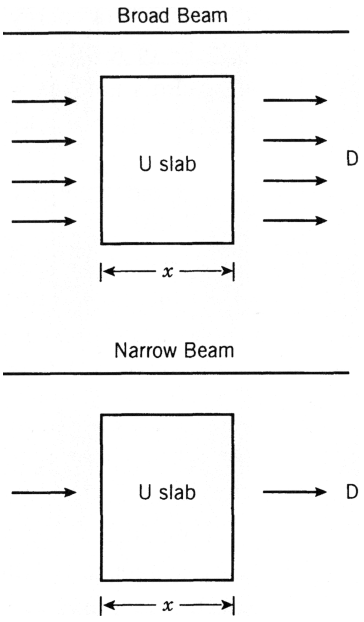

Figure 1.3 Geometry for broad and narrow photon beam scattering experiments. The detector position is indicated by the letter 'D'.

The measured radiation levels for three different thicknesses (x) are given below for both the broad-beam and narrow-beam situations. The following data are provided:

Density of uranium = 18.9 g/cm^3

Measured Radiation Levels for Various Thicknesses of Uranium.

Slab Thickness (cm)	Broad Beam (mR/hr)	Narrow Beam (mR/hr)
0.0	127.0	127.0
1.0	43.1	29.5
2.0	13.0	7.7
3.0	4.0	1.9

From these data, determine:

1.90 The linear attenuation coefficient of uranium for the narrow beam of 1-MeV photons.
1.91 The buildup factor for the broad beam with a slab thickness of 2.5 cm. Assume the linear attenuation coefficient is 2/cm.
1.92 What is the mass-attenuation coefficient of uranium for 1-MeV photons if the linear attenuation coefficient is 2/cm?

Scenario 1.24

The International Commission on Radiological Protection (ICRP) publishes various reports on basic radiation protection policy, practices, and research. Two such reports, Report Number 23 and Report Number 26, are of interest in this question.

1.93 ICRP Report Number 23 describes reference man as containing 140 g of potassium. The following data apply:

Mass of reference man = 70 000 g

0.012% of the potassium is K-40

K-40 decays by emitting a beta particle with a 90% probability

Maximum beta energy = 1.3 MeV

Half-life of K-40 = 1.2×10^9 years

Avogadro's number = 6.023×10^{23}

1.6021×10^{-6} erg/MeV

What is the average beta dose rate in rad per week to the whole body from K-40?

1.94 Which statement is most accurate?
   a. It is hard to identify K-40 in the presence of 10 nCi of Co-60.
   b. The quantity of K-40 does not vary by more than ±5% from individual to individual.
   c. K-40 has no regulatory significance in the whole-body counting program but serves as an important qualitative system check.
   d. K-40 should be omitted from the radionuclide library for the whole-body counting because it is of no regulatory interest.
   e. A multidetector counter will typically not identify K-40.

1.95 Strict adherence to ICRP Report Number 26 would allow:
   a. plutonium internal doses to be regulated using annual dose equivalent rather than committed dose equivalent.
   b. deletion of record-keeping for internal doses less than 50% of the allowable dose limit.
   c. consideration of internal and external dose limits separately.
   d. the worker to choose the type of respiratory protection device if use is required.
   e. use of air samples and stay-time calculations instead of respirator usage, if it is deemed to be ALARA.

1.96 The assumption of electronic equilibrium for a Co-60 source at 1-m distance is least likely to be correct at the:
   a. surface of the skin.
   b. center of a large muscle mass.
   c. bone–tissue interface.
   d. center of a large bone mass.
   e. internal surface of the lung.

Scenario 1.25

A worker at your facility received a diagnostic administration of I-131 as NaI for a thyroid function test. Your radiation protection program restricts workers to 0.1 times the ICRP-10 investigation level from certain work even if the exposure resulted from a medical procedure.

You are asked to estimate how long he must be placed on a restricted status. He is also concerned about the dose that he will receive from this diagnostic procedure. Use the following data to answer the questions for this scenario:

Administered activity = 1.0 μCi

Administrative limit (0.1 times the ICRP-10 investigation level) = 30 nCi

Biological half-life (thyroid) = 74 days

Biological half-life (whole body) = 0.4 days

Physical half-life = 8.08 days

$f_2 = 0.3$

Thyroid mass = 20 g

$S(T \leftarrow S)$ for the thyroid = $2.2 \cdot 10^{-2}$ rad/μCi-hr (MIRD-11)

1.97 Calculate the thyroid dose received from this procedure.

1.98 Based on the thyroid retention, how many days must pass until the worker can be released from restricted status?

1.99 Which one of the following statements is incorrect?

a. Because a thyroid abnormality is suspected, these calculations are only an estimate of the organ dose.
b. The most accurate method of assessing the actual dose is to obtain *in vivo* bioassay data and calculate an organ retention function.
c. The values of *S* take beta dose within the organ of interest into account, but do not consider beta doses between organs except for organs with walls and bone and bone marrow.
d. Because of the reciprocal dose theorem, the dose to testes from the thyroid is equal to the dose to the thyroid that would be produced if the same activity were in the testes.
e. The ICRP-10-derived investigation level for short-lived transportable radionuclides is based on one-quarter of the maximum permissible quarterly intake for short-lived transportable radionuclides.

Scenario 1.26

You are responsible for the health physics input to the design of a new laboratory which will be handling tritium. One room in the lab is 30 ft long, 20 ft wide, and 10 ft high and contains the primary tritium handling glove box. The glove box will contain a maximum of 10 Ci of tritium, all of which could be released into the room should an accident occur. You are concerned about the doses that could be received by an operator in the room and by an individual standing downwind at the site boundary, 1 mile away. The following data should be considered:

Breathing rate of both individuals = 3.5×10^{-4} m^3/s

Atmospheric diffusion factor at 1 mile = 1.0×10^{-4} s/m^3

Dose conversion factor for tritium (including absorption through the skin) = 158 rem/Ci inhaled

Time the operator remains in the room after the accident without respiratory protection = 30 min

Assume that the position of the operator in the room does not affect the dose received.

1.100 Determine the maximum dose equivalent that could be delivered to the operator.

1.101 Calculate the maximum dose equivalent that could be delivered to the person at the site boundary. Assume that all tritium is released to the environment in 30 min and that the person stays at the boundary for the entire release.

1.102 The ventilation system design criteria call for three complete air changes per hour in the lab. If the ventilation system works as designed, what is the maximum dose the operator could receive?

1.103 If the ventilation system works as designed, what is the maximum dose equivalent that could be delivered to the person at the site boundary?

1.104 Your Design goal for the dose equivalent delivered to the operator during the tritium accident is 500 mrem. How many air changes per hour will be required to limit the operator's dose equivalent to this value?

Scenario 1.27

1.105 Calculate the committed dose equivalent (CDE), the committed effective dose equivalent (CEDE), the annual limit on intake, and the derived air concentration for the inhalation of Cs-137. The Cs-137 decay scheme is illustrated in Figure 1.4.

The following data are given:

Specific Effective Energy (MeV per gram per transformation) of Cs-137.

Targets	Sources	
	Lungs	**Total Body**
Gonads	0.0	2.7×10^{-6}
Breast	0.0	2.7×10^{-6}
Red Marrow	0.0	2.7×10^{-6}
Lungs	1.9×10^{-4}	2.7×10^{-6}
Thyroid	0.0	2.7×10^{-6}
Bone surface	0.0	2.7×10^{-6}
SI wall	0.0	2.7×10^{-6}
ULI wall	0.0	2.7×10^{-6}
LLI wall	0.0	2.7×10^{-6}
Uterus	0.0	2.7×10^{-6}
Adrenals	0.0	2.7×10^{-6}

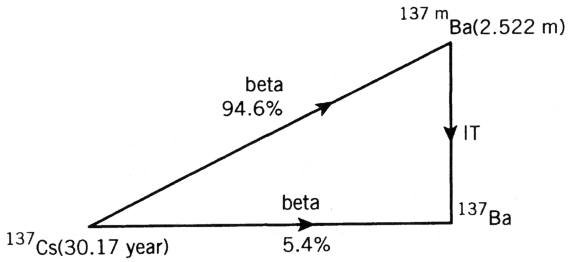

Figure 1.4 Decay scheme for Cs-137.

Specific Effective Energy (MeV per gram per transformation) of Ba-137m.

Targets	Sources	
	Lungs	Total Body
Gonads	5.7×10^{-8}	4.7×10^{-6}
Breast	2.7×10^{-6}	3.9×10^{-6}
Red Marrow	2.5×10^{-6}	4.3×10^{-6}
Lungs	9.5×10^{-5}	4.0×10^{-6}
Thyroid	2.6×10^{-6}	3.9×10^{-6}
Bone surface	2.0×10^{-6}	4.0×10^{-6}
SI wall	5.9×10^{-7}	4.9×10^{-6}
ULI wall	8.0×10^{-7}	4.8×10^{-6}
LLI wall	1.7×10^{-7}	4.9×10^{-6}
Uterus	2.1×10^{-7}	4.9×10^{-6}
Adrenals	4.9×10^{-6}	5.2×10^{-6}

Number of Nuclear Transformations Over 50 Years in Source Organs or Tissues per Unit Intake of Activity (Transformations/Bq) of Cs-137 (U_S).

Organ	Isotope	Oral	Inhalation (Class D)
	Cs-137	$f_1 = 1.0$	$f_1 = 1.0$
	Ba-137m	$f_1 = 1.0$	$f_1 = 1.0$
Lungs	Cs-137		1.9×10^4
	Ba-137m		1.8×10^4
Other tissue	Cs-137	1.2×10^7	7.7×10^6
(whole body 70 000 g)	Ba-137m	1.2×10^7	7.3×10^6

f_1 is the fraction of a stable element reaching the body fluids following ingestion.

Scenario 1.28

Assuming that Tc-99m acts as an insoluble compound, calculate the following for an uptake of 1 μCi of Tc-99m into the stomach:

1.106 The cumulated activity of Tc-99m in μCi-hr in each segment of the ICRP-30 gastrointestinal (GI) tract.
1.107 The dose equivalent in rem to the walls of each segment of the GI tract.
1.108 The maximum permissible uptake rate in μCi/hr and the maximum allowed concentration in water for occupational exposure in μCi/ml. Assume that an intake of 1100 ml/day of contaminated water is consumed during the work day and that the maximum allowed organ dose permitted at your facility is 15 rem.

ICRP-30 GI Tract Parameters.

Section of GI Tract (i)	Mass of walls (g)	Mass of Contents (g)	Mean Residence Time (day)	λ_i (day^{-1})
Stomach (ST)	150	250	1/24	24
Small intestine (SI)	640	400	4/24	6
Upper large intestine (ULI)	210	220	13/24	1.8
Lower large intestine (LLI)	160	135	24/24	1

Absorbed Dose per Unit Cumulated Activity (rad/μCi-hr) for Tc-99m with a Half-Life of 6.03 hr.

Target Organs	Source Organs			
	Stomach Contents	SI Contents	ULI Contents	LLI Contents
GI stomach wall	1.3×10^{-4}	3.7×10^{-6}	3.8×10^{-6}	1.8×10^{-6}
GI SI wall	2.7×10^{-6}	7.8×10^{-5}	1.7×10^{-5}	9.4×10^{-6}
GI ULI wall	3.5×10^{-6}	2.4×10^{-5}	1.3×10^{-4}	4.2×10^{-6}
GI LLI wall	1.2×10^{-6}	7.3×10^{-6}	3.2×10^{-6}	1.9×10^{-4}

Scenario 1.29

Two possible approaches for estimating the risk of cancer induction from exposure to low levels of ionizing radiation are ICRP-26 and the Probability of Causation (PC) Tables published by the Department of Health and Human Services (HHS).

Answer the following questions to demonstrate your understanding of these reports.

1.109 Based upon ICRP risk estimates, what is the probability of developing a radiation-induced fatal cancer over a lifetime for an average occupationally exposed radiation worker who has received 100 000 mrem of uniform, whole-body external exposure?

1.110 Assuming a normal cancer fatality rate of 20%, what would be the total probability of developing a fatal cancer for a group of occupationally exposed workers with a 3 in 1000 probability of contracting a radiation-induced fatal cancer?

1.111 The ICRP-26 risk model for cancer is based on:
- a. An absolute risk model.
- b. A relative risk model.
- c. An absolute and relative risk model.
- d. A stochastic model.
- e. A linear stochastic model.

1.112 The PC Tables are based on:
- a. An absolute risk model.
- b. A relative risk model.
- c. An absolute and relative risk model.
- d. A stochastic model.
- e. A linear stochastic model.

1.113 Which statement is not true regarding the PC tables?
- a. The formulation of these tables was mandated by Congress.
- b. Smoking history is not considered when using the tables to estimate risk.
- c. Resource of data for the table includes: rodent data, *in vitro* cell studies, and human data.
- d. The tables were published to provide scientific evidence to resolve radiation litigation cases.
- e. Prior medical X-ray exposure is not considered when using the tables to estimate risk.

Scenario 1.30

The biological effects of ionizing radiation encompass a broad range of topics. The following questions are designed to indicate your general understanding of this area.

1.114 Equal amounts of tritium as tritiated water and tritiated thymidine (a basic component of DNA) are incorporated into a large volume of cells. Which statement best describes the biological effectiveness of these compounds?
- a. Tritiated water will cause more biological damage to the cell because the cell is principally made up of water.

b. Tritiated water will cause more biological damage to the cell because tritiated thymidine is quickly metabolized by the cell.
c. Tritiated thymidine will cause a greater biological effect than tritiated water because it is incorporated into the cell's nucleus.
d. Both compounds will deliver the same biological effect because they are distributed in equal activities.
e. The biological effect will be the same for both compounds because both emit the same low-energy beta radiation and are equal in activity.

1.115 Match the following inhaled radionuclides with the adult critical organ. The critical organ may be used more than once.

a. Lung ___ Strontium-90 (soluble).
b. Bone ___ Cesium-137.
c. Total body ___ Plutonium-239 (soluble).
d. Liver ___ Uranium-238 (insoluble).
e. Kidney ___ Radon-222.

1.116 Based on the law of Bergonie and Tribondeau, order the following cells from most to least radiosensitive:

a. Mature lymphocytes.
b. Intestinal crypt cells.
c. Mature spermatocytes.
d. Erythrocytes (red blood cells).
e. Nerve cells.

1.117 Ionizing radiation has been directly associated with cataract formation. Select the statement that is incorrect.

a. The cataractogenic dose response is considered a threshold effect.
b. Fast neutrons are more effective at producing cataracts than are other forms of radiation.
c. The cataract effect is dependent on age at the time of irradiation.
d. Occupational exposure to X-rays accounts for approximately 1% of the cataracts observed in X-ray technicians.
e. Radiogenic cataracts are distinct in that they originate on the anterior epithelium of the lens.

1.118 Figure 1.5 can be used to express cell survival under a number of different irradiation circumstances. Which of the following statements is not true?

a. Curve A best represents the response of a cell system to a high dose rate, whereas curve B best represents the response to a low dose rate.
b. Curve B best represents a multitarget cell system response, whereas curve A best represents a single-target system.
c. Curve A best represents the effect in a cell system that is irradiated under hypoxic conditions, whereas curve B best represents the response of the same system under aerated conditions.
d. Curve B best represents the response of a cell system to low LET radiation, whereas curve A best represents the response of the same system to high LET radiation.

e. Curve B best represents the response of a cell system when a radioprotective compound is used, whereas curve A best represents the response of that cell system without a radioprotective compound.

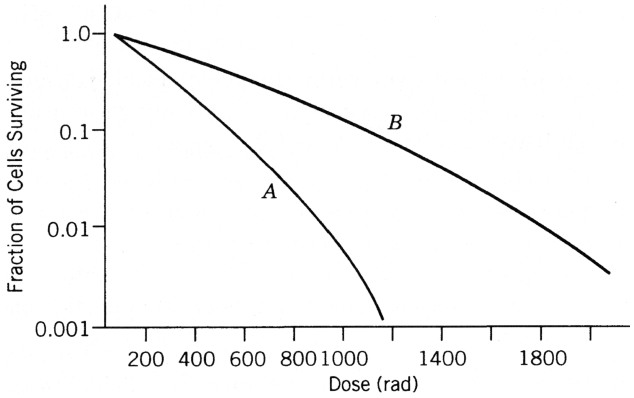

Figure 1.5 Fractional cell survival curves as a function of dose.

Scenario 1.31

The following series of questions relates to dosimetry and dose limits.

1.119 The major pathway by which soluble radioactive material is removed from the body is
   a. Perspiration.
   b. Feces.
   c. Respiration.
   d. Exhalation.
   e. Urine.

1.120 Prior to January 1993, the internal dose assessment methodology used to meet regulatory requirements in Title 10 CFR Part 20 is:
   a. NCRP-84.
   b. NCRP-91.
   c. ICRP-30.
   d. ICRP-2.
   e. ICRP-26.

1.121 The ICRP (ICRP-26) determined that, to prevent nonstochastic effects, the annual dose equivalent limit which must not be exceeded for all tissues except the lens of the eye is:
   a. 30.0 rem
   b. 5.0 rem
   c. 5.0 Sv
   d. 0.5 Sv
   e. 30 rad.

1.122 In order to limit stochastic effects, the dose limit recommended in ICRP-26 is based on the principle that the risk associated with uniform irradiation of the whole body is
a. equal to
b. greater than
c. less than
d. related to
e. not related to
the risk associated with nonuniform radiation.

1.123 ICRP-26 replaces the ICRP-2 concept of the critical organ with the concept of:
a. Genetic region.
b. Source region.
c. Organ equivalent.
d. Tissue region or target tissue.
e. Weighted critical organ.

1.124 Most large data sets of measurable occupational annual dose equivalents have been found to fit a:
a. Poisson distribution.
b. Normal distribution.
c. Log-normal distribution.
d. Binomial distribution.
e. Weibull distribution.

1.125 In 1980, the ICRP reviewed its annual dose limitation recommendations on the lens of the eye.
They decided to:
a. change its recommendations from 0.3 Sv to 0.15 Sv.
b. change its recommendations from 0.3 Sv to 0.50 Sv.
c. leave the number unchanged.
d. drop its recommendations for the lens of the eye.
e. make the eye limitations 0.50% of that for other tissues.

1.126 NCRP-116 recommends that the occupational cumulative effective dose limit should be:
a. 10 mSv × age (y).
b. 20 mSv × age (y).
c. 50% larger than the NCRP-91 value.
d. 50% smaller than the NCRP-91 value.
e. 50 mSv × (age (y) – 18).

1.127 Assuming an average annual dose equivalent of 5 mSv and employing BEIR III methodology, the annual risk estimate (total radiation-induced effects) for occupational radiation workers is considered to be a nominal value of about:
a. 1×10^{-2}
b. 1×10^{-3}
c. 1×10^{-4}

d. 1×10^{-5}
e. 1×10^{-6}.

How would your answer change if BEIR V and BEIR VII were the basis of the risk coefficient?

1.128 The average annual fatal accident rate in safe industries in the United States is approximately:
a. 1×10^{-3}
b. 1×10^{-4}
c. 1×10^{-5}
d. 1×10^{-6}
e. 1×10^{-7}.

Scenario 1.32

The biological effects of ionizing radiation depend upon the tissues involved and the nature of the radiation impinging upon the cellular structures. One of the more radioresilient tissues is the skin of the whole body. This scenario addresses radiation effects on skin.

1.129 Which one of the following, lists the skin response to acute radiation exposure in correct chronological order?
a. Dry desquamation, moist desquamation, erythema, recovery.
b. Moist desquamation, dry desquamation, erythema, recovery.
c. Dry desquamation, moist desquamation, recovery, erythema.
d. Erythema, moist desquamation, dry desquamation, recovery.
e. Erythema, dry desquamation, moist desquamation, recovery.

1.130 Which one of the following factors does not affect the severity of the skin's reaction to radiation?
a. Skin pigmentation.
b. Fractionation of dose.
c. Charged particle equilibrium at the basal cell layer.
d. Dose rate.
e. LET.

1.131 The radiosensitivity of skin is based on the sensitivity of which tissue?
a. Epidermal layer.
b. Basal cell layer.
c. Horny layer.
d. Hair follicle.
e. Fat cells.

1.132 If the skin were contaminated by an isotope with a half-life of 8 days, and assuming an exponential turnover time of the skin of 50% in 5 days, calculate the time to reduce the contaminant to 10% of the initial level. Assume that decontamination has been ineffective.

1.333 ICRP recommends a weighting factor of 0.01 for assessing stochastic risk to the skin.

This means that:

a. Radiogenic skin cancer is a low risk.
b. Radiogenic skin cancer is a high risk.
c. Spontaneous skin cancer is a low risk.
d. Radiogenic skin cancer exceeds spontaneous skin cancer as a risk.
e. Dose equivalent to the whole body is 100 times dose to the skin.

Scenario 1.33

The BEIR VII report gives an analysis for cancer risk assessment from exposure to low levels of ionizing radiation. This report was preceded by a number of studies that addressed human risk estimation. The following questions address biological risk from ionizing radiation.

1.134 There are several important areas for which human data are inadequate for risk estimation. Provide the types of information from animal studies that can be useful in human risk estimation.

1.135 Initiation, promotion, and progression are three distinct stages of experimental radiation-induced carcinogenesis. Identify the radiobiological factors which affect either the onset or the development of malignant tumors in experimental animals.

1.136 For this question, assume that all radiation induced effects lead to death. Based upon BEIR V, the weighted average risk of death following an acute dose equivalent of 0.1 Sv of low-LET radiation to all body organs is estimated to be:

a. 1×10^{-4}
b. 8×10^{-3}
c. 2×10^{-3}
d. 8×10^{-4}
e. 1×10^{-2}.

1.137 Two competing functional forms have been used for describing fatal cancer risks from radiation exposures.

Choose the best statement:

a. The BEIR V multiplicative risk model multiplies the dose by a constant to determine cancer risk.
b. The BEIR IV additive risk model adds a constant times the underlying risk of cancer to the age- and sex-dependent radiation dose to determine cancer risk.
c. The BEIR V relative risk model computes fatal cancer risk for individuals by an age- and sex-dependent factor times the cancer risk in the victim's relatives.
d. The additive risk model has been dropped by the BEIR V Committee in favor of the multiplicative risk model.
e. The BEIR IV absolute risk model holds that fatal cancer risk is a linear function of the absolute value of the radiation dose.

1.138 Consider a general form of a cancer risk estimate from BEIR V:

$$r(d) = r_0 \, [1 + f(d) \, g(B)]$$

where
$r(d)$ = total risk
r_0 = background risk
$f(d)$ = function depending on the dose d
$g(B)$ = function of dose-modifying parameters B.

Which of the following statements is not correct?
a. The constant 1 ensures positive values of the excess risk estimate.
b. $g(B)$ may include components which depend on sex and age.
c. $f(d)$ can be a linear or linear-quadratic function.
d. r_0 can vary significantly for different populations at risk.
e. r_0 is not specifically modeled by the Committee.

Scenario 1.34

You are employed as the Radiation Health Manager by the Big Pharma Corporation (BPC) at their Elephant's Ear, AZ production facility. BPC is licensed for radiopharmaceutical production using a variety of isotopes. The BPC license is based on ICRP-26 methodology.

A worker on the production line where radioiodine materials are manufactured appears to have a positive result for I-131 in a spot urine sample. The sample was taken late in the worker's shift, after the majority of the day was spent working with I-131. No air sample results are available, but thyroid counting capability exists. The chemical form of the I-131 is NaI.

From external thyroid counting, the following data are obtained:

Time Post Intake (d)	Thyroid Activity (kBq)	IRF[a, b, c]
1	250	0.133
7	230	0.0995
10	130	0.0751

[a] Intake Retention Function (IRF) for inhalation of Class D I-131.
[b] Fraction of intake expected to be in the thyroid at this time post-intake.
[c] Radioactive decay is included in these values.

For inhalation of Class D I-131, the dose conversion factor for the thyroid is 2.9×10^{-7} Sv Bq^{-1}. Approximately 75% of the I-131 as NaI is excreted from the body in the urine in 1–2 days with an effective halftime of about 6 hours. The remaining 25% of I-131 is trapped in the subject's thyroid, reaching a maximum about 24 hours post-intake, and is excreted with an effective halftime of about 7 days. The ICRP-26 organ weighting factor (w_T) for the thyroid is 0.03.

1.139 Given that you can choose *in vivo* or *in vitro* methods of analysis to perform bioassay, describe and discuss the optimal approach for this case. In your discussion list two advantages and two disadvantages for *in vitro* and *in vivo* methods of analysis as related to this case.

1.140 How might your approach to bioassay change as time goes by, given the metabolic model for iodine?

1.141 Based on the thyroid counting data, what is your best estimate of the subject's intake? For this question, use the NUREG/CR-4884 (1987) methodology.

1.142 Assume that the intake was 5 MBq. What is the committed dose equivalent (CDE) to the thyroid for this intake? What is the committed effective dose equivalent (CEDE) for this intake? Assume that organs other than the thyroid make a negligible contribution to the CEDE. Have any regulatory limits been exceeded?

Scenario 1.35

You are an assistant professor in the Nuclear and Radiological Sciences Department at the Le Chat Institute of Technology. The following questions will be part of an exam in HP303, External Radiation Dosimetry for undergraduate health physics students. This question involves a LiF thermoluminescent dosimeter and the ICRP-60 and NCRP-116 recommendations. Candidate dosimeter configurations follow.

Chip	Material	Thickness (cm)	Cover
1	^{7}LiF	0.38	100 mg/cm^2 copper and 200 mg/cm^2 plastic
2	^{7}LiF	0.38	1000 mg/cm^2 plastic
3	^{7}LiF	0.15	7 mg/cm^2 mylar
4	^{6}LiF	0.38	300 mg/cm^2 plastic
5	^{7}LiF	0.38	300 mg/cm^2 plastic
6	^{6}LiF	0.38	300 mg/cm^2 plastic and Cd filter

1.143 An employee works in a mixed radiation field, which includes beta particles, gamma photons, alpha particles, and thermal and mixed energy fast neutrons. The absorbed dose from external sources in the work environment was reported to be 30 μGy beta, 70 μGy gamma, 90 μGy thermal neutrons, and 25 μGy fast neutrons with an average energy of 10 MeV. Calculate the ICRP-60 equivalent dose in μSv.

1.144 A 26 year old radiation worker had a lifetime effective dose of 0.32 Sv. Compare this worker's lifetime dose to the recommendations of NCRP-116.

1.145 A radiation worker recorded the following effective doses over the past 4 years:

Year	Effective Dose (mSv)
1	10
2	30
3	40
4	20

According to ICRP-60, what is the maximum recommended dose for this worker in year 5?

1.146 Given the information in the problem statement, construct a dosimeter to measure the effective dose for a laboratory worker using a Pu/Be neutron source. State the number of chips to be included in the dosimeter and limit the number of chips to a maximum of four.

1.147 Given the information in the problem statement, construct a dosimeter to measure the effective dose for an X-ray technologist. State the number of chips to be included in the dosimeter and limit the number of chips to a maximum of four.

1.148 A portable meter (i.e., BF_3) could be used to determine the neutron effective dose to an individual with:

a. knowledge of the neutron spectrum so that the proper RBE can be determined.
b. knowledge of the relationship between the neutron energy spectrum and the energy of the neutron calibration source, the ratio of gamma and neutron fluence rates, and the individual's stay-time.
c. knowledge of the magnitude of the effective dose due to photons to be subtracted from the total effective dose (i.e., the meter is 'zeroed') and application of a neutron energy correction.
d. knowledge of how the instrument responds to the spectrum as compared to the neutron calibration source as well as the individual's stay-time.
e. knowledge of near laboratory conditions controlling temperature, humidity, neutron energy, and fluence rate.

Scenario 1.36

You have been retained by the Nuclear Regulatory Commission for developing a revision to 10CFR20, which will be based on the 1990 recommendations of the ICRP. These recommendations are contained in ICRP Publication 60, *1990 Recommendations of the International Commission on Radiological Protection*, and in subsequent ICRP publications.

The NRC has requested the answers to several questions in order to enhance their understanding of ICRP-60 and its supporting documents. A portion of these

questions involves an intake of I-131. The Annual Limit on Intake (inhalation) for I-131 is 1.0×10^6 Bq. The breathing rate of the average worker is 20 L min^{-1}, and the tissue weighting factor (w_T) for the thyroid is 0.05.

1.149 Define the following ICRP-60 dosimetric quantities: a. equivalent dose, b. effective dose and c. committed effective dose.
1.150 What are the ICRP-60 recommended occupational limits for effective dose?
1.151 For a declared pregnant worker, what is the ICRP-60 recommended limit for: a. external exposure and b. intake of radioactive material?
1.152 An individual performs a job in a room that contains airborne radioactive materials. The room concentration of I-131 is 8.3×10^4 Bq/m^3. The job requires 30 minutes to complete. Calculate the committed effective dose to the worker.
1.153 If the individual in Question 1.152 suffers from thyroid disease (for example hyper- or hypo-thyroidism), why do you need additional information to evaluate the committed effective dose? What information do you need to enhance the dose assessment?

Scenario 1.37

Radiation litigation cases frequently require a technical assessment of the biological effects of ionizing radiation. You have been retained by the law firm of Whiplash, Ripov, and Scam to participate in the case preparation for a suit being filed against a nuclear utility for exposure to ionizing radiation during a worker's career. The worker was exposed to both Co-60 and fission neutrons during her career.

You have been requested to explain the dose response curves for chromosome aberrations in human lymphocytes exposed to Co-60 gamma rays (γ) and to fission spectrum neutrons (n). Following ICRU Report No. 40, the number of chromosome aberrations per cell in human lymphocytes (e) when exposed to fission neutrons is $e_n = 0.60\ D_n$ where D is the absorbed dose in Gy. For photons, the number of chromosome aberrations per cell in human lymphocytes when exposed to photon radiation is $e_\gamma = 0.0157\ D_\gamma + 0.05\ D_\gamma^2$.

1.154 What is the primary mode of interaction for the following radiation types in tissue: fast neutrons, thermal neutrons, and Co-60 gamma rays?
1.155 What property of the neutrons and gamma-rays accounts for the difference in shape of the two dose response relationships defined above?
1.156 What is the relative biological effectiveness (RBE) for neutrons for an effect of 0.5 chromosome aberrations per cell?
1.157 What is the maximum value of the RBE for chromosome aberrations for neutrons based on the information provided in the problem statement?
1.158 What value should be used for the quality factor Q for neutrons with unspecified energies? Provide the basis (source) for your answer.

Scenario 1.38

An inhalation incident involving airborne Co-60 and I-131 has occurred at the Alabama National Radiochemistry and Environmental Laboratory (ANREL). The worker immediately showered, changed clothing, and received a whole body count. Analysis of material in the incident area suggests that the Co-60 was a Class Y compound and the I-131 was Class D. Laboratory procedures specify the use of ICRP-26/30 methodology.

Co-60 has a half-life of 5.27 y, a stochastic inhalation ALI (Class Y) of 1×10^6 Bq, and a committed dose equivalent (CDE) in the lungs of 3.4×10^{-7} Sv/Bq. The ($f_{N\text{-}P}, f_{T\text{-}B}, f_P$) values are (0.0, 0.0, 100.0) where $f_{N\text{-}P}, f_{T\text{-}B}, f_P$ are the percentage contributions of the CDE in the three lung regions due to initial depositions in the nasopharyngeal (N-P), tracheobronchial (T-B), and pulmonary (P) lung regions, respectively.

Fractions of Initial Intake Remaining in the Whole Body as a Function of Inhaled Particle Size (μm) and Elapsed Time.

Elapsed Time (d)	Inhaled Particle Size		
	1 μm	5 μm	10 μm
0	0.63	0.91	1.00
1	0.57	0.80	0.87
5	0.18	0.10	0.09
10	0.14	0.06	0.04
15	0.13	0.05	0.04
20	0.12	0.05	0.03

ICRP-26 Recommended Weighting Factors.

Organ or Tissue	w_T
Gonads	0.25
Breast	0.15
Red Bone Marrow	0.12
Lung	0.12
Thyroid	0.03
Bone Surfaces	0.03
Remainder	0.30

Correction for particle size:

$$\frac{H_{50}\ (\text{AMAD})}{H_{50}\ (1\ \mu\text{m})} = f_{\text{N-P}}\ \frac{D_{\text{N-P}}\ (\text{AMAD})}{D_{\text{N-P}}\ (1\ \mu\text{m})} + f_{\text{T-B}}\ \frac{D_{\text{T-B}}\ (\text{AMAD})}{D_{\text{T-B}}\ (1\ \mu\text{m})} + f_{\text{P}}\ \frac{D_{\text{P}}\ (\text{AMAD})}{D_{\text{P}}\ (1\ \mu\text{m})}$$

Fraction of Intake Deposited in the Lung Compartments.

Particle Size AMAD (μm)	Deposition Probabilities			
	$D_{\text{N-P}}$	$D_{\text{T-B}}$	D_{P}	Sum
1	0.30	0.08	0.25	0.63
5	0.74	0.08	0.09	0.91
10	0.87	0.08	0.05	1.00

1.159 The Co-60 component of the whole body count result was 7.77×10^5 Bq. Assuming that the activity median aerodynamic diameter (AMAD) of the aerosol was 1 μm, estimate the intake, expressed in percent ALI, based on the whole body count result.

1.160 Calculate the committed effective dose equivalent (CEDE) for an inhalation intake of 9.25×10^5 Bq of 1 μm AMAD Class Y Co-60.

1.161 For this part only, assume that the CEDE due to Co-60 was 0.5 mSv (50 mrem). The worker has an I-131 intake that resulted in 6 mSv (600 mrem) committed dose equivalent (CDE) to the thyroid. Assume that the thyroid is the only significantly irradiated organ or tissue. During the same monitoring period, the worker also received 2.5 mSv (250 mrem) due to external radiation exposure from Co-60. What is the total effective dose equivalent (TEDE) to the worker during the monitoring period?

1.162 Another worker inhaled 1.11×10^6 Bq of Class Y Co-60. The AMAD was determined to be 10 μm. Calculate the committed dose equivalent (CDE) to the lungs.

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