AMERICAN BOARD OF HEALTH PHYSICS

EXAMINATION 32, PART II

July 8,1988

FUNDAMENTALS (Answer all 6 questions in this section)

QUESTION 1

Two possible approaches for estimating the risk of cancer induction from exposure to low levels of ionizing radiation are:

Report #26, published by the International Commission on Radiation Protection (ICRP), and

The Probability of Causation (PC) Tables published by the Department of Health and Human Services (HHS).

Select the single best correct answer to each of the following questions.

POINTS

1 A. Based on the 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 external exposure?

1. 1 in 5
2. 1 in 10
3. 1 in 100
4. 1 in 1,000
5. 1 in 10,000

1 B. 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 1,000 probability of contracting a radiation-induced fatal cancer?

1. 3 in 1,000
2. 20 in 1,000
3. 230 in 1,000
4. 203 in 1,000
5. 200 in 1,000

Points

1 C. The ICRP Risk Model for cancer is based on:

1. an absolute risk model.
2. a relative risk model.
3. an absolute and relative risk model.
4. a stochastic model.
5. a linear stochastic model.

1 D. The Probability of Causation Tables are based on:

1. an absolute risk model.
2. a relative risk model.
3. an absolute and relative risk model.
4. a stochastic model.
5. a linear stochastic model.

1 E. Which statement is not true regarding the PC Tables?

1. The formulation of these tables was mandated by Congress.
2. Smoking history is not considered when using the tables to estimate risk.
3. Sources of data for the table include: rodent data, in vitro cell studies and human data.
4. The tables were published to provide scientific evidence to resolve radiation litigation cases.
5. Prior medical x-ray exposure is not considered when using the tables to estimate risk.

QUESTION 2

ANSI N13.11 - 1983, "American National Standard for Dosimetry-Personnel 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:

POINTS

1 A. The provisions of this standard apply:

1. to neither pocket dosimeters norextremity dosimeters.
2. to pocket dosimeters but not to extremity dosimeters.
3. only to beta and gamma radiation.
4. to extremity dosimeters but not to pocket dosimeters.
5. to film badges but not to TLDs.

1 B. 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 doses received. For which one of the following tissues is this difference most significant:

1. red bone marrow.
2. skin.
3. gonads.
4. lens of the eye.
5. whole body.

Points

1 C. Because of the particular radiation sources specified, the standard least adequately tests for radiations emitted by:

1. C-14, power plant leakage neutrons
2. P-32, Cf-252
3. Y-90/Sr-90, Am-Be source
4. Co-60, Ni-65
5. Uranium slab, Cf-252

1 D. A dosimeter of a processor which has passed the test category for:

1. beta radiation is appropriate for measuring low energy photons.
2. beta radiation is not appropriate for measuring beta radiation from all beta sources.
3. low energy photons can be used to pass the performance test for beta radiation.
4. 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.
5. neutrons is appropriate for measuring neutron radiation from any source.

1 E. This standard:

1. forms the basis for the National Voluntary Laboratory Accreditation Program for dosimetry processors.
2. provides guidance for individual variability from reference man.
3. provides guidance for summing the external and internal dose.
4. is applicable to the entire range of gamma energies.
5. is not required to be implemented by 10 CFR Part 20.

QUESTION 3

For each of the situations below, 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/cm^2 mylar window.
2. A TLD albedo containing both ^6Li and ^7Li elements.
3. A TLD albedo containing both ^6Li and ^{11}B 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.

POINTS

1 A. An accelerator facility using tritiated targets with 14 MeV deuteron beams.

1 B. A mixed neutron and gamma field where gamma dose predominates.

1 C. A radiographer using a 320 kVp X-ray machine.

1 D. A field of high energy (6 MeV) photons.

1 E. A field of mixed beta (average energy of 200 keV) and gamma (average energy of 800 keV) radiation.

QUESTION 4

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

POINTS

2 A. ICRP Report #23 describes reference man as containing 140 grams of potassium. Given:

Reference man = 70,000 grams

0.012% of the potassium is ^{40}K

^{40}K decays by emitting a beta particle with 90% probability

Maximum beta energy is 1.3 MeV

Half Life of $^{40}K = 1.2 \times 10^9$ years

Avogadro's number = 6.023×10^{23}

1.6021×10^{-6} erg/MeV

What is the average beta dose rate in rad/week to the whole body?

1. 2.5×10^{-2}
2. 2.3×10^{-3}
3. 2.5×10^{-4}
4. 7.5×10^{-4}
5. 2.5×10^{-5}

1 B. Which statement is most accurate?

1. It is hard to identify ^{40}K in the presence of 10 nCi of ^{60}Co.
2. The quantity of ^{40}K does not vary by more than plus or minus 5% from individual to individual.
3. ^{40}K has no regulatory significance in a whole body counting program but serves as an important qualitative system check.
4. ^{40}K should be omitted from the radionuclide library for whole body counting since it is of no regulatory interest.
5. A multi-detector counter will typically not identify ^{40}K.

<u>QUESTION 4 (continued)</u>

1 C. Strict adherence to ICRP Report Number 26 would allow:

1. plutonium internal doses to be regulated using annual dose equivalent rather than committed dose equivalent.
2. deletion of record keeping for internal doses less than 50% of the allowable dose limit.
3. consideration of internal and external dose limits separately.
4. the worker to choose the type of respiratory protection device if use is required.
5. use of air samples and stay-time calculations instead of respirator usage, if it is deemed be ALARA.

1 D. The assumption of electronic equilibrium for a ^{60}Co source at one meter distance is <u>least</u> likely to be correct at the:

1. surface of the skin.
2. center of a large muscle mass.
3. bone/tissue interface.
4. center of a large bone mass.
5. internal surface of the lung.

QUESTION 5

A common type of portable beta-gamma survey instrument uses an air ionization chamber vented to atmospheric pressure. The cylindrical detector is 3 inches high and 3 inches 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 questions below with respect to the instrument's response versus the "true" dose rates specifically associated with the following conditions.

POINTS

POINTS		
1	A.	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	B.	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	C.	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	D.	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	E.	Briefly describe a source of error associated with measuring beta dose rates from high energy beta sources using open minus closed window readings.

QUESTION 6

Broad and narrow beams of 1-MeV photons are normally incident on different thicknesses of uranium slabs, as shown in Figure 1. The measured radiation levels for three different thicknesses (x) of the slab are given in Table 1, for both the broad and narrow beam situations. From the data given, determine:

POINTS		
2	A.	The linear attenuation coefficient of uranium for the narrow beam of 1 MeV photons.
2	B.	The buildup factor for the broad beam with a slab thickness of 2.5 cm. (Assume the linear attenuation coefficient is 2 cm^{-1})
1	C.	What is the mass-attenuation coefficient of uranium for 1 MeV photons if the linear attenuation coefficient is 2 cm^{-1}.

DATA

The density of uranium is 18.9 g/cm^3.

Figure 1

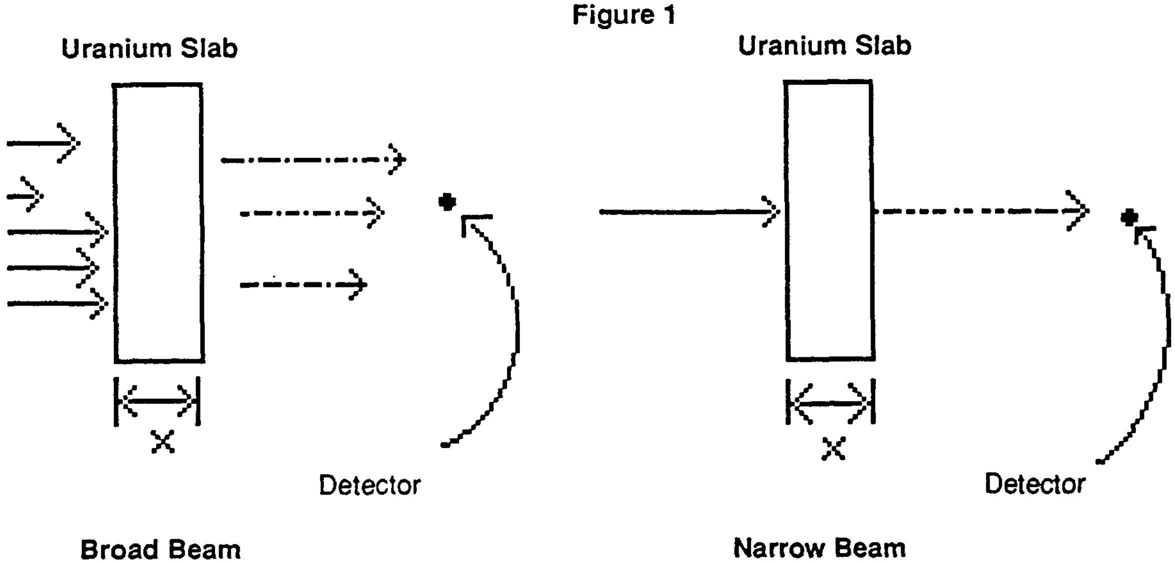


Table 1
Measured Radiation Levels for Various
Thicknesses of Uranium

Slab Thickness x (cm)	Broad Beam (mR/hr)	Narrow Beam (mR/hr)
0.0	127	127
1.0	43.1	29.5
2.0	13.0	7.7
3.0	4.0	1.9

AMERICAN BOARD OF HEALTH PHYSICS

EXAMINATION 32, PART II

COMPREHENSIVE CERTIFICATION

July 8, 1988

SPECIALTY (Answer any four of the specialty questions in this section)

QUESTION 7

You are the health physicist at a 4 MeV, 200mA, electron linear accelerator used for experimental and testing purposes in an industrial setting.

POINTS

1.25 A. An aluminum-walled water-filled box is used in the beam as a beam monitor and stopper. It was noted that the 1.78 MeV gamma ray of the 3 minute half-life ^{28}Al was seen on a spectrometer being used in the target area. What is the most likely explanation for this?

1. neutron activation of ^{27}Al resulting from gamma-neutron reaction in the beam.
2. neutron activation of ^{28}Al resulting from a gamma-neutron reaction with deuterium in the beam stopper.
3. isomeric transition of ^{28}Al, initiated by a gamma-neutron reaction with tritium present in the beam stopper.
4. neutron activation of ^{27}Al resulting from gamma-neutron reaction with deuterium in the beam stopper.
5. electron/positron pair formation with subsequent release of gamma ray from ^{28}Al.

QUESTION 7 (continued)

POINTS

1.25 B. The energy of the beam after the first 90 degree scatter of a 4 MeV bremsstrahlung beam is represented by which one of the following equations:

1. $E = T_e + T_p + 2\, m_o c^2$

2. $$E = \frac{E_o m_o c^2 (1 - \cos\theta)}{m_o c^2 + E_o}$$

3. $E = E_o m_o c^2 (1 - \cos\theta)$

4. $E = E_o m_o c^2 (\sin\theta)$

5. $$E = \frac{E_o m_o c^2}{m_o c^2 + E_o (1 - \cos\theta)}$$

Where:
E = Energy of the scattered beam
E_o = Initial energy of the beam
T_e = Kinetic energy of the electron
T_p = Kinetic energy of the positron
m_o = initial mass of the electron
c = speed of light
θ = scattering angle

1.25 C. One experimenter at your facility directs an electron beam into a copper target. The beam has been running for four hours using a water-cooled magnet. The water coolant is stopped and the accelerator scrams. The experimenter wants to rush in to fix his set-up. You, as health physicist assess the primary hazard to be:

1. ^{17}O.
2. ^{16}N.
3. There is no radiation hazard which would prevent the researcher from taking care of his experiment.
4. activated dust.
5. residual scatter.

QUESTION 7 (continued)

POINTS

1.25 D. Another experimenter wants to test ceramic materials under high beam power. The ceramic material consists of beryllium oxide. This is the first operation of this type at your facility and you must make a presentation to the accelerator experiment committee for approval of the protocol and authorization to use the accelerator.

The following controls are appropriate EXCEPT:

1. initial neutron surveys at low power.
2. activation surveys at high power.
3. thermal cooling time.
4. particulate respiratory protection.
5. high energy neutron personnel monitoring badges.

1.25 E. Your accelerator experiment committee is thinking about modifying its existing facility to house a 40 MeV linear electron accelerator, capable of a 2 Amp peak current, 0.5 microsecond pulse duration, and 250 pps. In evaluating the shielding design, which of the following would you NOT take into consideration:

1. physical configuration of beam and occupied areas.
2. gamma ray attenuation coefficients
3. neutron cross sections
4. probability and consequence of gamma- neutron reactions.
5. electron capture cross sections.

QUESTION 7 (Continued)

POINTS

1.25 F. If both the old machine and the new machine had equal dose rate outputs and lead were used as the shielding material, which machine would require less lead to shield the bremsstrahlung?

1. The 4 MeV machine because 4 MeV electrons are easier to shield.
2. The 4 MeV machine because 4 MeV gammas are easier to shield.
3. The 40 MeV machine because the 40 MeV electrons are easier to shield.
4. The 40 MeV machine because 40 MeV gammas are easier to shield.
5. The requirements are essentially the same because the broad beam TLV for the 4 Mev machine is nearly the same as that for the 40 MeV machine.

1.25 G. What is the qualitative relative importance of the neutron source for the two machines?

1. There is essentially no neutron production with the 4 MeV machine and significant neutron production with the 40 MeV machine.
2. The is some neutron production with the 4 MeV machine and essentially none with the 40 MeV machine.
3. The neutron production with the two machines is approximately the same.
4. Both machines produce significant neutrons, with considerably more neutron production with the 4 MeV machine than with the 40 MeV machine.
5. Neutron production with both machines can be neglected.

QUESTION 7 (Continued)

POINTS

1.25 H. What is the qualitative relative importance of the bremsstrahlung production as compared to the neutron production for the two machines if the shielding were constructed of concrete?

1. The bremsstrahlung production is sufficiently high for both machines to control the shielding design.
2. The bremsstrahlung production controls shielding design for the 4 MeV machine, but neutron production controls shielding design of the 40 MeV machine.
3. The bremsstrahlung production controls shielding design for the 40 MeV machine, but neutron production controls shielding design for the 4 MeV machine.
4. The neutron production is sufficiently high for both machines to control shielding designs.
5. Both neutron production and bremsstrahlung production control shielding thickness equally for both machines.

QUESTION 8

You are the health physicist for an engineering firm that is designing a new low-level radioactive waste disposal facility which will operate for 30 years. The Compact Region which needs this facility has chosen the site and has decided against using traditional shallow land disposal technology (used at Barnwell, Beatty, and Hanford). The Compact Commission wants to use an enhanced disposal technology in hopes of lessening public opposition to the facility. Your firm is to provide comprehensive licensing and design services for 3 conceptual designs for the Compact Commission.

The three designs are for:
1) above ground vaults, 2) earth-mounded concrete bunkers with large above ground concrete canisters for Class A waste and vaults below ground for Class B&C waste, and 3) below ground vault disposal for all classes of waste. Vaults are to be constructed of reinforced concrete and all waste is to be grouted in place. The concrete canisters are 6 feet in diameter and over 7 feet high (large enough to contain large high integrity containers and liners). The waste is also to be grouted in place in the canisters.

Preliminary dose estimates have been made for the three options and are presented in Table 1.

POINTS

POINTS		
7	A.	Outline a generic pre-operational environmental monitoring plan for the site chosen, including media to be sampled, locations, sample frequencies and type, and analyses to be performed.
1	B.	Identify the four major pathways of exposure which could be expected during the institutional care period.
1	C.	Discuss any potential health effects you might expect during the post-closure phase of each facility.
1	D.	The Compact Commissioner has reviewed the 3 designs, and asks you which disposal technology you recommend. As a health physicist, which option do you recommend to the Commissioner and why? Assume all regulatory requirements can be met by each of the three designs.

SEE NEXT PAGE FOR DATA

QUESTION 8 (continued)

DATA

TABLE 1

DOSE ESTIMATES

Scenarios	Disposal Options		
	below ground	above/below ground	above ground
Average worker dose in person-rem/year	3	3	5
Number of occupationally exposed employees	30	30	23
Maximum surface dose rate (mrem/yr) 100 years post closure	7.7×10^{-7}	4.9×10^{-7}	1
Committed dose equivalent to maximally exposed individual (mrem/year), 500 years post-closure.	1.5	1.3	4
Average population committed dose equivalent from 100 to 500 years post closure (person-rem/year).	10	9	30

QUESTION 9

PART I

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

Atom	Delta (MeV)
^{126}Te	-90.05
^{126}I	-87.90
^{126}Xe	-89.15

The energy equivalent of the electron rest mass is 0.511 MeV. The binding energy of the K-shell electron in Iodine is 32 keV.

For the following questions choose the single best answer.

Points

1 A. The energy released (Q-value) by the decay of ^{126}I via capture of a K-shell electron, going directly to the ground state of ^{126}Te, is:

1. 0.03 MeV.
2. 1.13 MeV.
3. 2.12 Mev.
4. 2.15 MeV.
5. 2.18 MeV.

1 B. The energy released (Q-value) by the decay of ^{126}I via positron emission to the ground state of ^{126}Te, is:

1. 0.51 MeV.
2. 1.02 MeV.
3. 1.13 MeV.
4. 1.64 MeV.
5. 2.15 MeV.

QUESTION 9 (Continued)

POINTS

1 C. The energy released in the decay of ^{126}I to the ground state of ^{126}Xe by B^- emission is:

1. 0.20 MeV.
2. 0.23 MeV.
3. 0.90 MeV.
4. 0.74 MeV.
5. 1.25 MeV.

1 D. Of the following kinds of radiation emitted from ^{126}I, which is the single least significant potential contributor to dose?

1. annihilation photons
2. bremsstrahlung
3. internal-conversion electrons
4. Auger electrons
5. antineutrino

1 E. Why are the 32 keV Te x-rays present with an ^{126}I source?

1. The nucleus of ^{126}Te has excess energy after the EC event. This excess energy is released by ^{126}Te as x-rays.
2. Stable ^{126}Te has excess energy after the B^+ emission. This excess energy is released by ^{126}Te as x-rays.
3. Electrons rearranging between the L and M shells produce x-rays.
4. Te x-rays are released when the EC event creates a vacancy in the inner shells and electrons from outer shells fill the vacancy.
5. Te x-rays are equivalent to the bremstrahlung radiation emitted by ^{126}I.

PART II

The nuclide ^{90}Sr (atomic number 38) decays (B^-) into ^{90}Y (atomic number 39), which then decays (B^-) into ^{90}Zr (atomic number 40), with the indicated half-lives as shown:

$$^{90}Sr \xrightarrow[27.7\ yr]{} {}^{90}Y \xrightarrow[64.2\ hr]{} {}^{90}Zr$$

Points

1 F. What is the mean, or average, life of a ^{90}Y atom?

1. 31.1 hours
2. 44.5 hours
3. 77.04 hours
4. 92.6 hours
5. 128.4 hours

2 G. What is the specific activity of ^{90}Y in SI units?

1. 5.42×10^{5} Bq/kg
2. 7.22×10^{16} Bq/kg
3. 2.01×10^{19} Bq/kg
4. 7.22×10^{19} Bq/kg
5. 6.49×10^{21} Bq/kg

2 H. Starting with a pure ^{90}Sr sample at time T = 0, a researcher finds that the ^{90}Y activity is 3.4 mCi at t = 72.0 hours. What was the activity of the ^{90}Sr at T = 0?

1. 1.84 mCi
2. 3.4 mCi
3. 4.37 mCi
4. 6.29 mCi
5. 7.39 mCi

QUESTION 10

You have been selected as the Health Physics Manager for the atomic vapor laser isotope separation (AVLIS) facility at Moose River. The facility represents USDOE's most advanced technology for the enrichment of uranium. The AVLIS technology is based on the selective photo-excitation and subsequent ^{235}U ionization of the uranium vapor by laser irradiation. The photo-ions formed by the interaction of the laser beam and uranium vapor are extracted from the vapor stream by the combined action of magnetic and electric fields. Enriched and depleted streams are collected on separate surfaces. The process unit is a large vacuum chamber into which solid uranium feed and laser light are admitted. The internal surfaces of the process unit, particularly those that contact uranium, will be removed periodically from the vacuum chamber for cleaning and refurbishment. The uranium vapor is generated by the impingement of an electron beam upon the surface of the solid uranium source.

Use your experience with components similar to those described above to answer the following questions. Clearly state all assumptions about the process equipment and their corresponding parameters. DATA REQUIRED TO ANSWER THIS QUESTION ARE GIVEN ON THE FOLLOWING PAGES.

POINTS		
2	A.	List 4 ionizing radiation hazards and their sources, 2 nonionizing radiation hazards and their sources, and 2 other physical hazards and their sources which can occur in this facility.
2	B.	List 6 engineered design features which could be included in the AVLIS facility to reduce the ionizing radiation hazards.
6	C.	Assume the facility has been operating for 6 months. In the process of removing enriched material from the unit, a technician violates standard operating procedures and stacks enriched uranium into a critical geometry. After some other procedural violations, a criticality results in which 1×10^{16} fissions occur. What is the approximate dose to the technician (in rad) who is standing behind a 4 inch polyethylene shield 20 feet from the source?

QUESTION 10 (Continued)

DATA

The lasers utilize 30,000 volt power supplies.

The AVLIS technology has the potential to produce uranium enriched to 90%

The electron beam and associated components utilize 50,000 volt power supplies.

The cleaning of the vacuum chamber internals will be performed manually.

Internal components will be contaminated with uranium dust.

The electromagnetic field strength is high enough to warrant consideration.

The intensity of the laser radiation warrants personnel protection.

Assume each fission from a criticality would produce 8 prompt gammas with an average energy of 1 MeV and 2.5 prompt neutrons with an average energy of 2 MeV.

Ignore delayed radiation from fission products.

Assume the average energy of the neutrons and gamma rays are representative of the entire spectrum.

The density of polyethylene is 1.4 g/cm^3.

Energy fluence rate, fluence rate, and specific gamma ray constant as a function of photon energy are given in Figure 10-1.

Mass attenuation coefficients are given in Table 10-1.

Attenuation of fast-neutron dose is presented in Figure 10-2.

Flux to dose conversion factors are presented in Table 10-2.

TABLE 10-1

MASS ATTENUATION COEFFICIENTS--Continued

Photon Energy	SiO_2	NaI	Air	Con-crete	0.8N H_2SO_4	Bone	Muscle	Poly-styrene	Lucite	Poly-ethyl-ene	Bake-lite	Pyrex Glass
keV												
10	19.0	139.	4.99	26.9	5.76	20.3	5.27	2.13	3.25	2.01	2.76	17.1
15	5.73	47.4	1.55	8.24	1.76	6.32	1.63	0.755	1.06	0.728	0.923	5.14
20	2.49	22.3	0.752	3.59	0.849	2.79	0.793	.424	0.551	.420	.492	2.25
30	0.859	7.45*	.349	1.19	.391	0.962	.373	.259	.298	.266	.277	0.786
40	.463	19.3	.248	0.605	.276	.512	.268	.217	.234	.226	.223	.431
50	.318	10.7	.208	.392	.231	.349	.227	.199	.208	.209	.200	.302
60	.252	6.62	.188	.295	.208	.274	.205	.188	.193	.198	.187	.242
80	.194	3.12	.167	.213	.185	.209	.183	.173	.176	.183	.171	.190
100	.169	1.72	.154	.179	.171	.180	.170	.163	.164	.172	.161	.166
150	.140	0.625	.136	.144	.150	.149	.149	.145	.146	.154	.143	.139
200	.126	.334	.123	.127	.137	.133	.136	.132	.133	.140	.130	.125
300	.108	.167	.107	.108	.118	.114	.118	.115	.115	.122	.113	.107
400	.0959	.117	.0954	.0963	.106	.102	.105	.103	.103	.109	.101	.0954
500	.0874	.0955	.0870	.0877	.0965	.0927	.0960	.0938	.0941	.0995	.0921	.0870
600	.0808	.0826	.0805	.0810	.0893	.0857	.0888	.0868	.0871	.0921	.0852	.0804
800	.0707	.0676	.0707	.0709	.0783	.0752	.0779	.0763	.0765	.0809	.0749	.0704
MeV												
1.0	.0636	.0586	.0636	.0637	.0704	.0676	.0700	.0685	.0687	.0727	.0673	.0633
1.5	.0518	.0469	.0518	.0519	.0573	.0550	.0570	.0558	.0559	.0592	.0548	.0516
2.0	.0447	.0413	.0445	.0448	.0492	.0473	.0489	.0478	.0480	.0507	.0470	.0444
3.0	.0363	.0366	.0358	.0365	.0396	.0383	.0393	.0383	.0385	.0405	.0377	.0361
4	.0317	.0351	.0308	.0319	.0340	.0331	.0337	.0327	.0329	.0345	.0322	.0314
5	.0287	.0346	.0275	.0290	.0303	.0297	.0300	.0290	.0292	.0305	.0286	.0282
6	.0266	.0347	.0252	.0270	.0277	.0274	.0274	.0263	.0266	.0277	.0260	.0263
8	.0241	.0355	.0223	.0245	.0243	.0244	.0240	.0228	.0232	.0239	.0227	.0237
10	.0226	.0368	.0204	.0231	.0222	.0226	.0219	.0206	.0211	.0215	.0206	.0222
15	.0209	.0402	.0181	.0215	.0194	.0204	.0192	.0176	.0182	.0182	.0178	.0204
20	.0203	.0433	.0170	.0210	.0182	.0194	.0179	.0162	.0168	.0166	.0164	.0198
30	.0202	.0484	.0162	.0210	.0172	.0189	.0168	.0149	.0157	.0151	.0153	.0195
40	.0204	.0520	.0161	.0213	.0169	.0189	.0165	.0144	.0153	.0145	.0148	.0198
50	.0208	.0548	.0161	.0218	.0168	.0190	.0164	.0142	.0151	.0142	.0147	.0201
60	.0212	.0571	.0162	.0222	.0169	.0193	.0165	.0142	.0151	.0141	.0147	.0204
80	.0218	.0605	.0165	.0229	.0171	.0197	.0167	.0142	.0152	.0141	.0148	.0210
100	.0224	.0629	.0168	.0235	.0174	.0201	.0170	.0144	.0154	.0142	.0150	.0215
150	.0234	.0670	.0174	.0247	.0180	.0210	.0175	.0147	.0159	.0145	.0154	.0225
200	.0241	.0695	.0179	.0254	.0184	.0215	.0179	.0150	.0162	.0147	.0157	.0232
300	.0250	.0724	.0185	.0264	.0190	.0223	.0185	.0155	.0167	.0152	.0162	.0240
400	.0256	.0741	.0189	.0269	.0194	.0228	.0189	.0158	.0171	.0155	.0166	.0245
500	.0260	.0752	.0192	.0273	.0197	.0231	.0192	.0160	.0173	.0157	.0168	.0249
600	.0262	.0760	.0194	.0276	.0199	.0233	.0194	.0162	.0175	.0159	.0170	.0252
800	.0266	.0771	.0197	.0281	.0202	.0237	.0197	.0165	.0178	.0161	.0173	.0256
GeV												
1	.0269	.0778	.0199	.0283	.0204	.0239	.0199	.0166	.0180	.0163	.0174	.0258
1.5	.0273	.0789	.0202	.0287	.0207	.0243	.0202	.0169	.0182	.0165	.0177	.0262
2	.0275	.0794	.0204	.0290	.0209	.0245	.0203	.0171	.0184	.0167	.0179	.0264
3	.0278	.0800	.0206	.0292	.0211	.0247	.0205	.0173	.0186	.0169	.0181	.0267
4	.0279	.0803	.0207	.0294	.0212	.0249	.0206	.0174	.0187	.0170	.0182	.0268
5	.0280	.0805	.0208	.0295	.0213	.0249	.0207	.0174	.0188	.0170	.0183	.0269
6	.0281	.0807	.0208	.0295	.0213	.0250	.0208	.0175	.0188	.0171	.0183	.0269
8	.0281	.0808	.0209	.0296	.0214	.0251	.0208	.0175	.0189	.0172	.0184	.0270
10	.0282	.0809	.0209	.0297	.0214	.0251	.0209	.0176	.0189	.0172	.0184	.0271
15	.0283	.0811	.0210	.0298	.0215	.0252	.0209	.0176	.0190	.0173	.0185	.0271
20	.0283	.0812	.0210	.0298	.0215	.0252	.0210	.0177	.0190	.0173	.0185	.0272
30	.0283	.0813	.0211	.0298	.0216	.0253	.0210	.0177	.0191	.0173	.0185	.0272
40	.0284	.0813	.0211	.0299	.0216	.0253	.0210	.0177	.0191	.0173	.0186	.0272
50	.0284	.0814	.0211	.0299	.0216	.0253	.0210	.0177	.0191	.0173	.0186	.0272
60	.0284	.0814	.0211	.0299	.0216	.0253	.0210	.0177	.0191	.0174	.0186	.0273
80	.0284	.0815	.0211	.0299	.0216	.0253	.0210	.0178	.0191	.0174	.0186	.0273
100	.0284	.0815	.0211	.0299	.0216	.0253	.0211	.0178	.0191	.0174	.0186	.0273

* K edge of Iodine--33.2keV 5.69, 30.9.

Table 10-2

FLUX TO DOSE CONVERSION FACTORS

Neutron Energy (MeV)	Flux to Dose Conversion Factor (rad/neutron/cm^2)
2.5×10^{-8} (thermal)	5.1×10^{-10}
1×10^{-7}	5.1×10^{-10}
1×10^{-6}	6.2×10^{-10}
1×10^{-5}	6.2×10^{-10}
1×10^{-4}	6.0×10^{-10}
1×10^{-3}	5.1×10^{-10}
1×10^{-2}	4.0×10^{-10}
1×10^{-1}	8.0×10^{-10}
5×10^{-1}	2.3×10^{-9}
1	3.3×10^{-9}
2.5	3.9×10^{-9}
5	5.4×10^{-9}
7	5.8×10^{-9}
10	6.3×10^{-9}
14	7.7×10^{-9}
20	7.9×10^{-9}
40	9.9×10^{-9}
60	1.2×10^{-8}
1×10^{2}	1.2×10^{-8}
2×10^{2}	1.5×10^{-8}
3×10^{2}	1.8×10^{-8}
4×10^{2}	2.0×10^{-8}

Figure 10-1

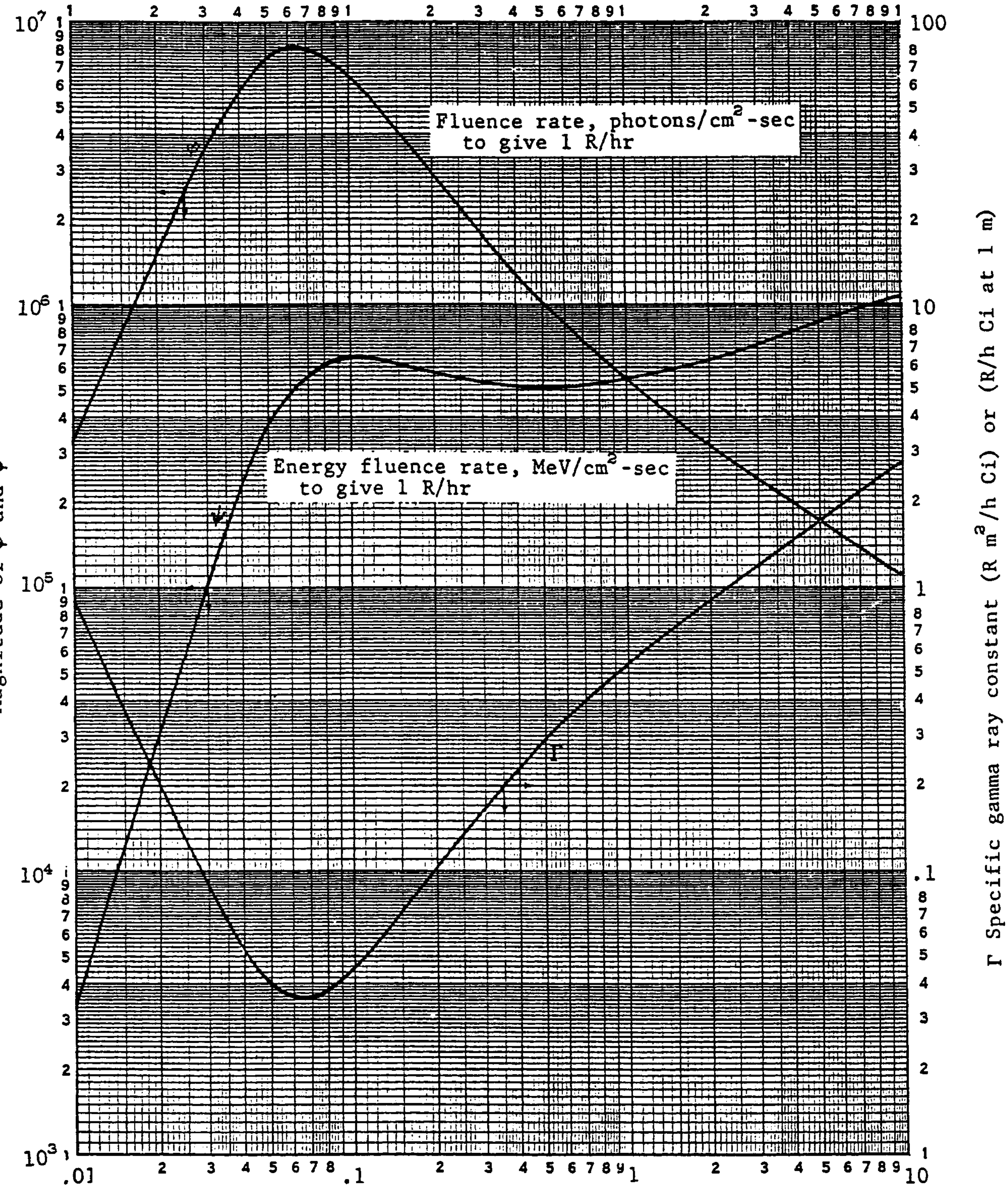


FIGURE 10-1

Page 25

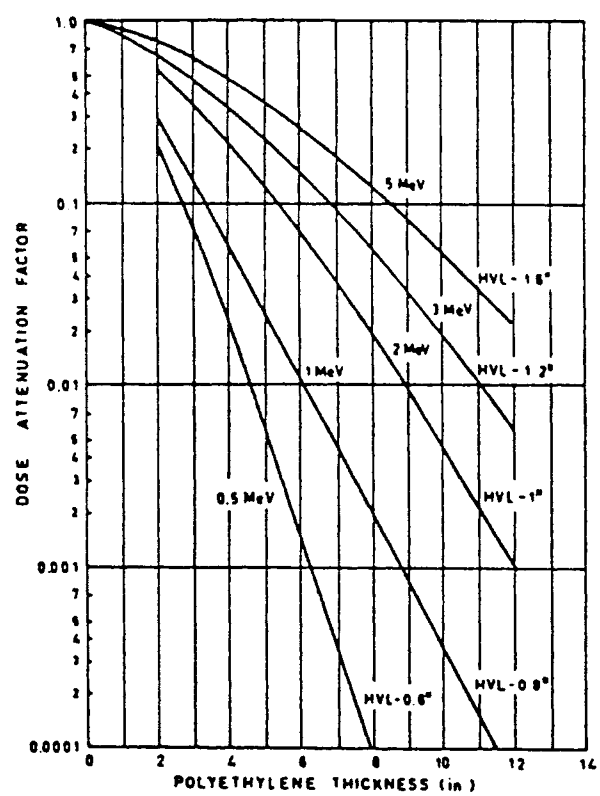
FIGURE 10-2

ATTENUATION OF FAST-NEUTRON DOSE

The attenuation factor for a polyethylene shield is given
The approximate equivalent thicknesses of other materials relative to polyethylene are:

1 in. polyethylene = 1 in. paraffin
= 1.07 in. water
= 2.1 in. concrete
= 2.7 in. soil saturated with water
= 3.75 in. dry soil

The equivalences for concrete and soil are approximations and vary with the concrete or soil composition.



Attenuation of fast-neutron dose.

QUESTION 11

Your hospital has decided to dedicate a lead shielded room for brachytherapy patients to reduce exposure from this treatment to staff, adjacent patients and visitors. Typical treatments for gynecological implants are ^{137}Cs sealed sources at a maximum treatment time of 50 hours each. Assume the adjacent patient could remain in her bed for an entire 7 day period during her hospital stay. Also assume that the brachytherapy room is continuously occupied for at least 100 hours in a week by a patient implanted with ^{137}Cs. A diagram is provided as Figure 11-1. Tables showing gamma ray transmission through lead and concrete are provided as Figures 11-2 and 11-3, respectively. DATA REQUIRED TO ANSWER THIS QUESTION ARE GIVEN ON THE FOLLOWING PAGES.

POINTS

2 A. Based on a maximum activity of 70 mg Ra equivalent, calculate the amount of lead in wall A necessary to comply with NCRP #37 recommendations for adjacent patient exposure per visit. Assume the dose at the wall controls implementation of the NCRP recommendations.

3 B. Using the same information as in the preceding question, calculate how much concrete is needed for the ceiling above.

1 C. List three differences related to radiation protection for ^{137}Cs and ^{125}I brachytherapies.

2 D. A patient who received typical ^{125}I brachytherapy using seeds to treat prostate cancer had a confirmed ^{125}I thyroid burden of 300 nanocuries detected 3 days after the procedure. Calculate the dose equivalent from the measured uptake to the thyroid using ICRP-26 metholology. Ignore back-decay.

2 E. A week after the initial 300 nanocurie measurement, the measured uptake is now 330 nanocuries. As a health physicist, how would you evaluate this situation?

QUESTION 11 (Continued)

DATA

Specific Gamma Ray Constant for Radium = 8.25 mR-ft^2/mg-hr

Patient attenuation = 30%

Brachytherapy patient is: 2 feet from wall A
(REFER TO FIGURE 11-1) 4 feet from adjacent patient's bed
7 feet from a nursery above
able to move about the room

HVL for ^{137}Cs in lead = 0.65 cm

Mass attenuation coefficient for concrete = 0.06 cm^2/g

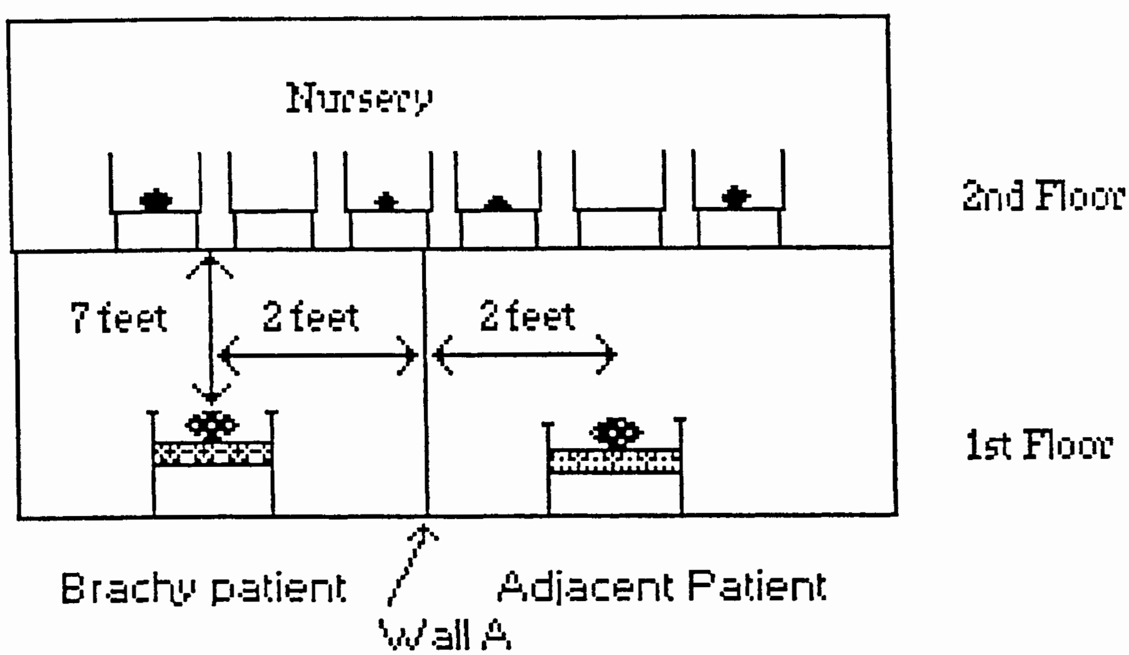
Density of concrete = 2.4 g/cm^3

Physical half-life of ^{125}I = 60 days

Biological half-life of ^{125}I = 138 days

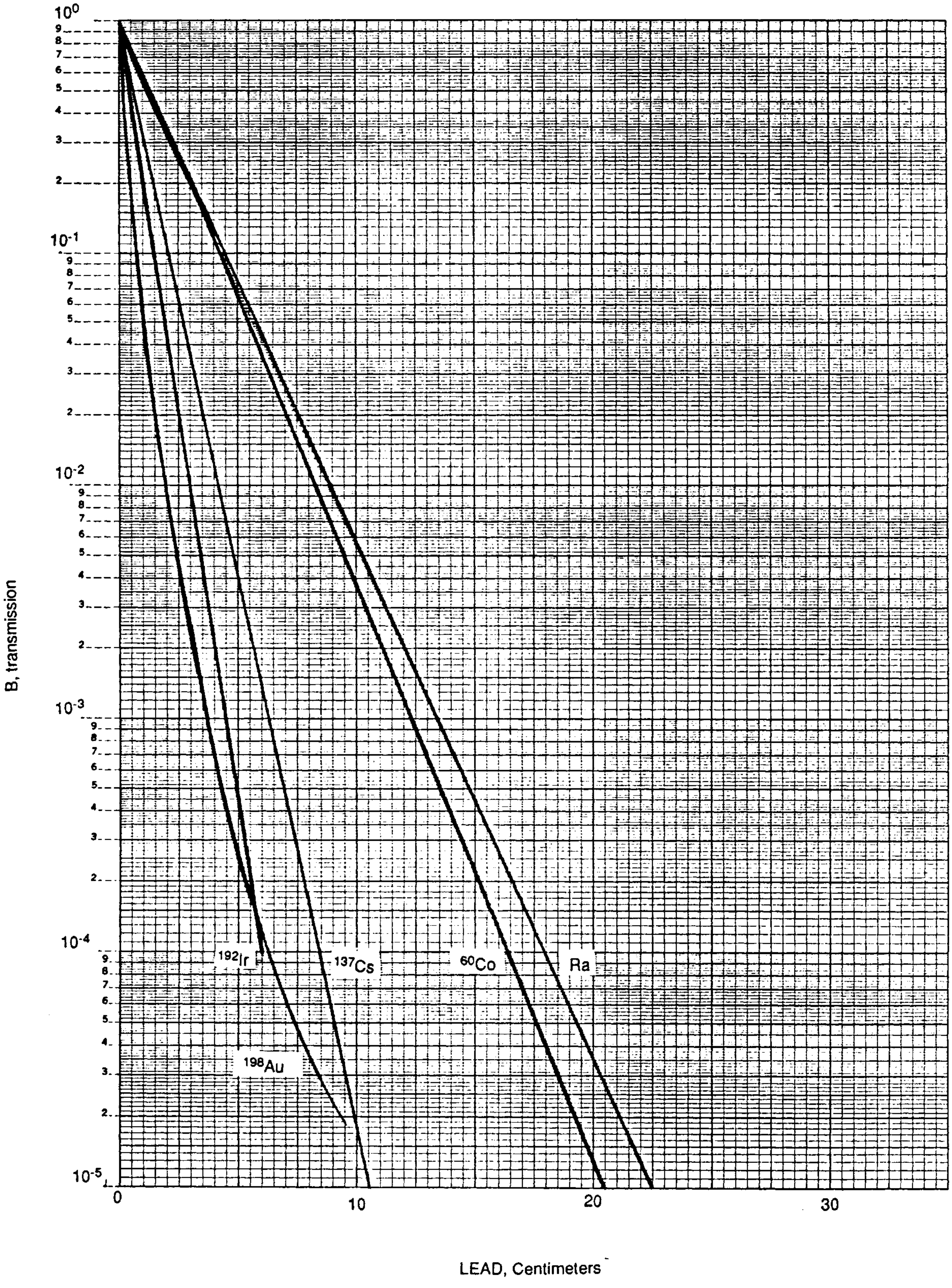
S for ^{125}I = 3 x 10^{-3} rad/uCi-hr

Figure 11-1



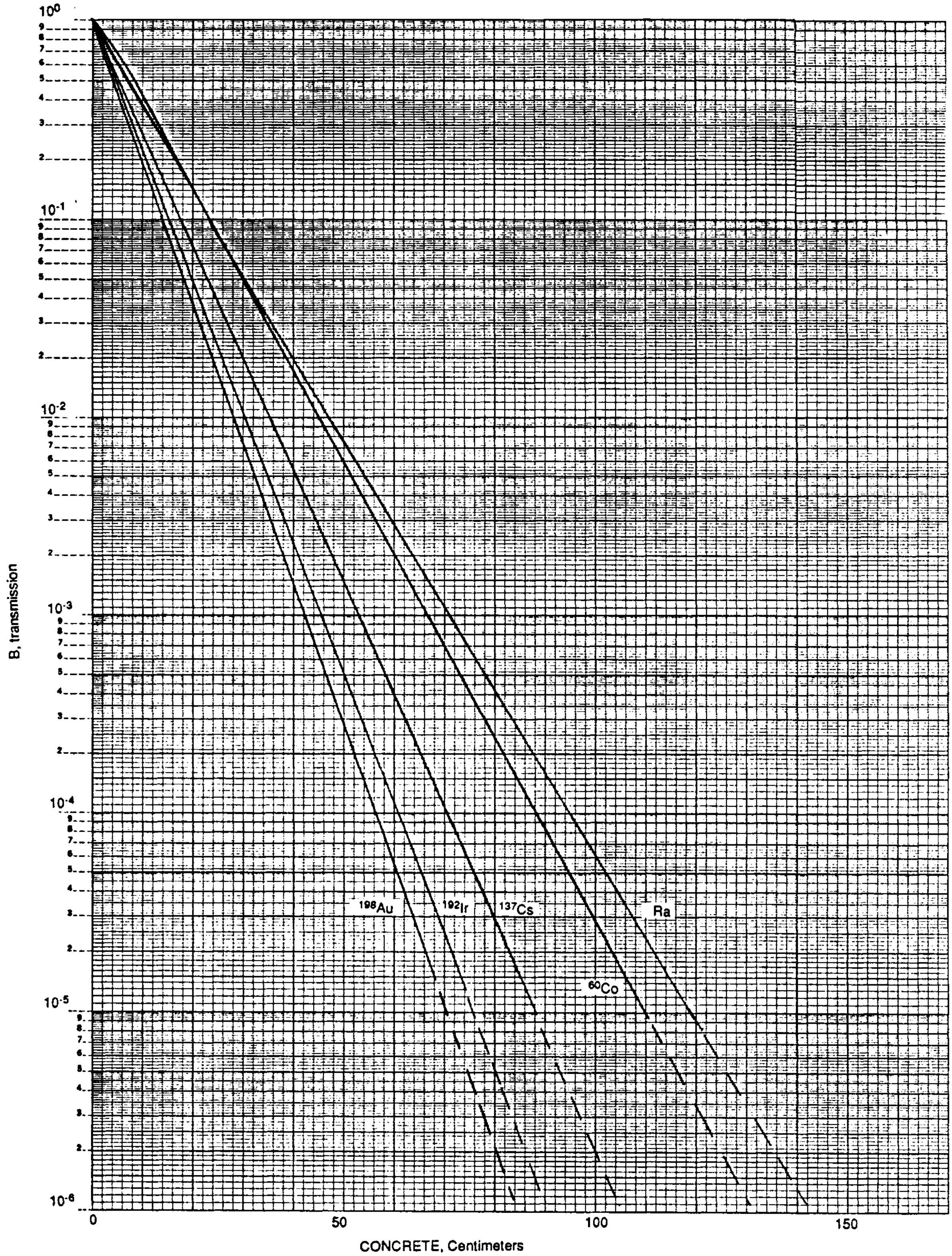
CROSS-SECTION OF HOSPITAL

FIGURE 11-2



Transmission through lead of gamma rays from selected radionuclides. Radium (Wyckoff and Kennedy [23]); cobalt-60, cesium-137, gold-198 (Kirn *et al.* [13]); iridium-192 (Ritz [24]).

FIGURE 11-3



Transmission through concrete, density 2.35 g cm⁻³ (147 lb ft⁻³), of gamma rays from selected radionuclides.

QUESTION 12

As a health physicist at a power reactor, you have been assigned the lead role in evaluating "hot particle" contamination of station personnel. "Hot particles" are very small highly radioactive particles with high specific activity. These particles have recently been detected with increasing frequency at your facility. Isotopic analyses indicate the particles are composed of fuel or neutron-activated corrosion and wear products.

On Tuesday morning you receive a phone call about a hot particle which was removed from under a pipefitters fingernail. Upon reporting for work, the pipefitter alarmed the entrance portal contamination monitor. The contamination was detected during a follow-up frisk, and was succesfully removed on the first attempt with sticky tape. Although not visible to the naked eye, the particle was analyzed for its constituent radionuclides.

DATA

An analysis of the contamination on the sticky tape led to the following results:

Radionuclide	Activity (uCi)	Dose Factor **(Note 1)** (rad-cm^2/uCi hr)
^{60}Co	3.9×10^{-1}	4.13
^{65}Zn	1.0×10^{-6} (LLD)	0.106
^{95}Zr	1.0×10^{-6} (LLD)	4.87
^{95}Nb	1.0×10^{-6} (LLD)	0.865

Note 1: Beta dose factors applicable for skin dose calculations averaged over 1 cm^2.

Based upon a review of the event, you determine the following:

Time of contamination:	1545 on Monday
Time of removal:	0730 on Tuesday (following day)
Source of contamination:	unknown, although the worker's protective clothing is strongly suggested

QUESTION 12 (continued)

Using the above data and your experience choose the single best answer to the following questions:

POINTS

1 A. In general, for a hot particle on the skin, the principal contributor to the dose to the skin in the vicinity of the particle is:

1. the neutron radiation emitted from the particle.
2. the gamma radiation emitted from the particle.
3. the alpha radiation emitted from the particle.
4. the conversion electron emitted from the particle.
5. the beta radiation emitted from the particle.

1 B. The most plausible explanation for the increasing frequency of detection of "hot particles" at nuclear power plants is:

1. the increased use of more sensitive instrumentation for detecting and measuring contamination by these particles.
2. an increase in the rate of production of these particles at nuclear power plants.
3. the changes in plant chemistry which have enhanced fuel reliability but have increased the corrosion rate of other components.
4. a decrease in the average corrosion particle size as plants have aged.
5. the trend toward increasing the time between refueling outages with a subsequent decrease in preventive maintenance.

1 C. Prior to 1988, explicit recommendations on limits for radiation exposure of skin by hot particles were provided by:

1. the ICRP, but not the NCRP.
2. both the NCRP and the ICRP.
3. the NCRP but not the ICRP.
4. neither the NCRP nor the ICRP.
5. the ICRU, NCRP, and ICRP.

QUESTION 12 (Continued)

POINTS

1 D. Radiobiological evidence suggests that, when compared to more uniform irradiation by the same quantity of radioactive material, highly localized beta irradiation of skin, such as from a particle on the skin:

1. is less likely to cause skin cancer.
2. is more likely to cause skin cancer.
3. is about equally likely to cause skin cancer.
4. is likely to cause an erythema within a few hours.
5. is likely to cause a small necrotic lesion on the skin of an individual after only a few hours exposure.

1 E. For a typical beta-gamma survey meter with a 5 cm pancake probe, what approximate instrument efficiency is appropriate for a 2.0 cm diameter stainless steel disk of ^{90}Sr if the source-detector distance is 1 cm?

1. 1%
2. 5%
3. 35%
4. 50%
5 65%

2 F. The beta dose equivalent to the skin of the worker's finger is:

1. 1.6 rem
2. 8 rem
3. 25 rem
4. 33 rem
5. 33,000 rem

QUESTION 12 (Continued)

POINTS

1 G. For this scenario, assume the calculated total dose for regulatory purposes, beta and gamma, was 50 rem. Which one of the following statements best describes compliance with U.S. Nuclear Regulatory Commission regulations?

1. No limit was exceeded. All tissues received less than the yearly allowable regulatory limit of 75 rem for the extremities.
2. The whole body exposure was less than the quarterly allowable regulatory limit of 3 rem.
3. The skin of the whole body limit of 15 rem was exceeded.
4. The dose limit of 18.75 rem for the extremities was exceeded.
5. The annual whole body dose limit of 5 rem was exceeded.

1 H. Which one of the listed follow-up actions would not be appropriate if you calculated a 50 rem total dose for this scenario?

1. Survey of the worker's home, car, girlfriend's home, and the local restaurant where he ate the previous evening.
2. Review, recalibrate and evaluate the station's laundry monitoring systems.
3. Interview the worker to advise him of his rights under federal law and to answer questions about biological effects.
4. Evaluate sensitivity of portal monitors and other contamination detection instrumentation.
5 Thyroid monitoring and increased urinalysis frequency.

1 I. Which one of the following detectors would provide the best sensitivity for detecting these particles at the exit station?

1. Energy compensated G-M probe
2. Air proportional detector
3. Zinc sulfide scintillation probe
4. Intrinsic germanium crystal with single channel analyzer
5. Ionization chamber

QUESTION 13

A laboratory contains hoods to provide graduate students safe areas to use radiochemicals. Unknown to laboratory personnel, the motor serving the exhaust fan for the hoods is turned off for repair for one hour. During this time one grad student completes a dual labeling experiment using 10 millicuries of ^{32}P (as orthophosphate) and 10 millicuries of tritium (as tritium oxide) in a hood. Another grad student completes a procedure with 5 millicuries of ^{125}I (as sodium iodide solution). As soon as the hood condition is recognized the RSO is called by the grad students.

DATA

The laboratory is 20 feet x 20 feet x 10 feet

1 cubic foot = 28.3 liters

Breathing rate = 20 liters/minute

POINTS

POINTS			
	A.		As the RSO, you want to sample the air of the laboratory.
1		1.	What is the airborne potential for these three radionuclides?
1		2.	Describe how you will perform air sampling for potential airborne radionuclides.
1		3.	Justify whether any air sampling procedure should be isokinetic or anisokinetic.
	B.		Assume a maximum credible case: all potentially airborne radionuclides have volatilized or evaporated and are uniformly distributed in the room. Air exchange rate is low.
1		1.	Calculate the maximum uptake the grad students could have received.
1		2.	How much radioactivity could be in the critical organs?
1	C.		As the RSO, you perform urinalyses on the 2 students for appropriate sampling periods. Explain what analytical techniques would be used to assess radioactivity in the urine.
1	D.		If sodium borotritide was used instead of tritium oxide, how could you assess airborne potential?
1	E.		Describe 2 measures that should be instituted to prevent a recurrence of this accident.
2	F.		List 4 conditions that would affect the amount of radioactivity in the students.