

SOLUTIONS AND ANSWERS TO 1996 ABHP EXAM

QUESTION 1

GIVEN: ICRP risk coefficients (Publication 60) have been increased from a nominal fatality probability coefficient of $1.25 \times 10^{-2}/\text{Sv}$ to $4 \times 10^{-2}/\text{Sv}$ for a worker population and to a value **R** of $5 \times 10^{-2}/\text{Sv}$ for members of the general public.

SOLUTIONS AND ANSWERS(*):

- * A. Three reasons for the increase in the nominal fatality probability (risk) coefficient include:
1. revised dosimetry of Japanese bomb survivors (neutron dose now insignificant);
 2. the change from an additive risk projection model, which postulates broadly that the excess mortality is independent of the natural mortality, to a multiplicative risk projection model, which assumes that there is a simple proportion between the natural cancer mortality and the excess due to radiation for the whole time after a minimum latency period (This model change yields about a two fold increase in the calculated fatal cancer probability.); and
 3. an observed increase in total cancers, other than leukemia, in the Japanese study, still rising, largely as a result of the excess mortality in those exposed as children, which increases the probability values up to about 2 times depending on the tissue site and allowance made for neutron RBE.
- * B. Based on current risk coefficients, three factors that significantly contribute to the uncertainty of a specific risk calculation include:
1. uncertainty associated with the model used to describe how the risk varies as a function of time for persons exposed at various ages;
 2. uncertainty associated with the model used to describe how the risk increases with dose; and
 3. the statistical uncertainty associated with the relatively small number of radiation induced cases observed in a population in a given category compared to the normal incidence.
- C. The following equation is commonly referred to as a relative risk model:
- * A. $\gamma(\mathbf{d}) = \gamma_0 [1 + \mathbf{f}(\mathbf{d}) \mathbf{g}(\boldsymbol{\beta})]$, where:
- $\gamma(\mathbf{d})$ is the risk of death as a function of radiation dose equivalent \mathbf{d} in Sv ;
- γ_0 is the age-specific background risk of death due to a specific cancer;
- $\mathbf{f}(\mathbf{d})$ is a linear or linear-quadratic function of dose; and
- $\mathbf{g}(\boldsymbol{\beta})$ is an excess risk function that depends on gender, age, etc.
- D. Based upon a linear, non-threshold risk model and population lifetime risk **R** of 0.05 Sv^{-1} , the

SOLUTIONS AND ANSWERS TO 1996 ABHP EXAM

expected number **N** of fatal cancers is estimated for a population **P** of **125,000,000** people using natural gas over 1 year who receive an annual average effective dose equivalent **H_E** of **4x10⁻⁶ Sv**:

*
$$N = P H_E R = 25.$$

- * E. The organ that has the highest probability of radiation induced cancer as a result of exposure to radon progeny is the lungs.

SOLUTIONS AND ANSWERS TO 1996 ABHP EXAM

QUESTION 2

GIVEN: count of smear:

- C_{s+b} \equiv gross counts observed for smear = **1,000 c**;
- T_{s+b} \equiv counting interval for smear = **10 min**;
- C_b \equiv background count = **2,340 c**; and
- T_b \equiv background counting interval = **60 min**.

SOLUTIONS AND ANSWERS(*):

A. The net counting rate R_s and associated theoretical, propagated, Poisson standard error¹ $\hat{\sigma}_s$ are calculated:

*
$$R_s = R_{s+b} - R_b = \frac{1,000 \text{ c}}{10 \text{ min}} - \frac{2,340 \text{ c}}{60 \text{ min}} = 100 \text{ cpm} - 39 \text{ cpm} = \mathbf{61.0 \text{ cpm}}, \text{ and}$$

*
$$\hat{\sigma}_s = \left(\frac{R_{s+b}}{T_{s+b}} + \frac{R_b}{T_b} \right)^{1/2} = \left(\frac{100 \text{ cpm}}{10 \text{ min}} + \frac{39 \text{ cpm}}{60 \text{ min}} \right)^{1/2} = \mathbf{3.26 \text{ cpm}}.$$

B. The standard deviation $\hat{\sigma}_{s+b}$ (i.e. theoretical Poisson standard error) and *relative probable error*, $0.6745 \hat{\sigma}_{s+b}/C_{s+b}$, in the **count** C_{s+b} of the smear are calculated:

*
$$\hat{\sigma}_{s+b} = (C_{s+b})^{1/2} = (1,000)^{1/2} = \mathbf{31.6}, \text{ and}$$

*
$$0.6745 \frac{\hat{\sigma}_{s+b}}{C_{s+b}} = 0.6745 \left(\frac{31.6}{1,000} \right) = \mathbf{0.0213},$$

where - 0.6745 to + 0.6745 is the range of the normal deviate corresponding to a probability of 0.5 or 50%.

¹The hat symbol, ^, shown on $\hat{\sigma}_s$ is used to distinguish this Poisson, propagated point estimated value from the true population standard deviation σ_s in the net counting rate. The hat symbol is used on other sample statistics to distinguish them from the true population values.

SOLUTIONS AND ANSWERS TO 1996 ABHP EXAM

Comment: This question would make more sense if asked prior to part A and in terms of the *standard error* and *relative standard error* in the gross rate R_{s+b} rather than in the count C_{s+b} .

* C. The descriptions, A through F, are matched to the given statistical concepts, 1 through 5:

1. C, 2. A, 3. B, 4. E, and 5. D.

D. The optimum values for T_{s+b} and T_b given a total time T of **120 minutes** are calculated from the previous calculated value of **100 cpm** for R_{s+b} and **39 cpm** for R_b :

$$\frac{T_{s+b}}{T_b} = \left(\frac{R_{s+b}}{R_b} \right)^{1/2} = \left(\frac{100 \text{ cpm}}{39 \text{ cpm}} \right)^{1/2} = 1.60; \text{ so}$$

$$T_{s+b} = 1.60 T_b, \text{ and } 2.60 T_b = T = 120 \text{ min; so}$$

*
$$T_b = \frac{120 \text{ min}}{2.6} = \mathbf{46.2 \text{ min}}, \text{ and}$$

*
$$T_{s+b} = 120 - 46.2 = \mathbf{73.8 \text{ min}}.$$

SOLUTIONS AND ANSWERS TO 1996 ABHP EXAM

QUESTION 3

GIVEN: five traces (semilog plots of cpm vs. time in minutes), A through E, of the counting rate for a fixed filter constant air monitor (CAM), each representing a physically possible combination of a fixed or varying airborne concentration of a single radionuclide collected on the filter and starting at a steady state background level, increasing counting rate during the period of increased airborne activity, and then continuing after all *non-background* activity in the sampled air is gone.

Comment: The given information in the ABHP exam has been reworded for clarification. The background counting rate shown in the figure most likely represents counter background and a saturation filter activity from a steady state concentration of short lived radon progeny. Thus, the description of the trace at the end of the given ABHP statement: "and then continues after all activity in the sampled air is gone" is misleading because not **all** airborne activity is gone as shown by trace E where the counting rate returns to the natural background level.

SOLUTIONS AND ANSWERS(*):

- * A. The trace representing a short term puff release of a long lived radionuclide is Trace A (Answer A).
- * B. The trace representing the presence of a nuclide with a half-life of about 20 minutes is Trace D (Answer D).
- * C. The trace representing the presence of an exponentially increasing concentration of a long lived radionuclide is Trace C (Answer C), which is not intuitively obvious until proven for limiting assumptions.
- D. The average airborne concentration **U** for trace B is calculated from the given qualitative information in the problem, data shown in the figure, and the following:

- T** ≡ sampling interval = 248 min - 100 min = **148 min**;
- F** ≡ flow rate = 3 cfm = **$8.50 \times 10^4 \text{ cm}^3 \text{ min}^{-1}$** ;
- E** ≡ detector efficiency = **0.2 c d⁻¹**;
- R** ≡ filter retention = **0.98**; and
- R_s** ≡ net counting rate at end of sampling = 8,500 cpm - 50 cpm = **8,450 cpm**.

Based upon the assumption of no significant decay of the radionuclide during the sampling interval **T**, the average airborne activity concentration **U** is calculated:

*
$$U = \frac{(R_s)}{(R)(E)(F)(T)(2.22 \times 10^6 \text{ dpm } \mu\text{Ci}^{-1})} = 1.54 \times 10^{-9} \text{ } \mu\text{Ci cm}^{-3}.$$

SOLUTIONS AND ANSWERS TO 1996 ABHP EXAM

QUESTION 4

GIVEN: parameter values shown for photons of interest for the materials muscle (m) and bone (b) (Note: values correspond to a photon energy of about 1.25 MeV, which was not given in the problem.).

SOLUTIONS AND ANSWERS(*):

* A. Answers to the 4 parts are summarized:

1. Kerma **K** is defined as the sum ΔE_k of the initial kinetic energies of charged particles liberated by the interaction of indirectly ionizing radiation in a volume element of material divided by the mass Δm of that volume element as the mass approaches an infinitesimally small value. For monoenergetic photons, **K** can be calculated:

$$K \equiv \text{Limit}_{\Delta m \rightarrow 0} \left(\frac{\Delta E_k}{\Delta m} \right) = \Phi \frac{\mu}{\rho} E_{tr},$$

where Φ is the photon fluence at the point of interest.

2. Absorbed dose **D** is defined as the average energy ϵ absorbed in a volume element of material divided by the mass **m** of that material. For monoenergetic photons under conditions specified in 3 below, it may be calculated:

$$D \equiv \frac{\epsilon}{m} = \Phi E_\gamma \frac{\mu_{ab}}{\rho} = \Phi \frac{\mu}{\rho} E_{ab} \cong K = \Phi \frac{\mu}{\rho} E_{tr}.$$

where Φ is the photon fluence at the point of interest and E_γ is the photon energy.

3. Conditions that must exist at a point in a material for dose **D** to equal kerma **K** of photons (or other indirectly ionizing radiations) include: Radiative energy losses by liberated charged particles must be negligible (i.e., so E_{ab} equals E_{tr}). The fluence of indirectly ionizing radiation must be uniform in the region of space about this point at a depth within the material where energy spatial equilibrium may be assumed (i.e., where the kinetic energy of liberated charged particles is the energy absorbed). This energy spatial equilibrium condition requires for photons that their energy not be so large as to require the liberated electrons to travel mainly in the forward direction to conserve the original momentum of the photons, but their energy must be sufficiently large so that their mean free path greatly exceeds the ranges of all liberated electrons.
4. The purpose of the build-up cap on an ionization chamber is to assure an equilibrium distribution of electrons is present in the wall at the wall/gas interface so that this distribution of electrons produces the dose in the gas.

SOLUTIONS AND ANSWERS TO 1996 ABHP EXAM

- B. Answers are summarized to the three parts for a narrow beam of photons normally incident on a slab phantom consisting of 1 cm of muscle followed by 1 cm of bone where the kerma $\mathbf{K}(0)$ at the surface of the muscle is **100 J kg⁻¹**:

Note: In each part below, the contribution from any scattered photons is neglected, even back scattered photons.

1. The kerma $\mathbf{K}_m(\mathbf{x})$ in muscle at a depth \mathbf{x} of **1 cm** is calculated:

$$* \quad \mathbf{K}_m(\mathbf{x}) = \mathbf{K}_m(0) e^{-\mu_m \mathbf{x}} = (100 \text{ J kg}^{-1}) e^{-0.0626} = \mathbf{93.9 \text{ J kg}^{-1}}.$$

2. The factor \mathbf{F} representing the ratio $\mathbf{K}_b/\mathbf{K}_m$ of the kerma in the bone to that in muscle at the muscle/bone interface is calculated:

$$* \quad \mathbf{F} \equiv \frac{\mathbf{K}_b}{\mathbf{K}_m} = \frac{(\mu/\rho)_b}{(\mu/\rho)_m} = \frac{0.0604}{0.0626} = \mathbf{0.965},$$

because the \mathbf{E}_{tr} values are the same value of **0.588 MeV** for muscle and bone.

3. The ratio $\mathbf{K}_b(1 \text{ cm})/\mathbf{K}_b(0)$ of the kerma 1 cm deep in bone to that at 0 cm in bone is calculated:

$$* \quad \frac{\mathbf{K}_b(1 \text{ cm})}{\mathbf{K}_b(0)} = e^{-\mu_b (1 \text{ cm})} = e^{-\left(\frac{\mu}{\rho}\right)_b \rho_b (1 \text{ cm})} = e^{-(0.0604)(1.65)} = \mathbf{0.905}.$$

- * C. The depth \mathbf{x}_m where the largest absorbed dose occurs is obtained as follows. I assume that the radius of the photon beam is at least equal to the range of the most energetic electron produced in the muscle and that the incident photon beam is relatively free of charged particles (electrons). As photons in this narrow beam penetrate the muscle, they are attenuated, and through their interactions, produce a distribution of charged particles (electrons) that builds up below the surface. These electrons give rise to the dose where they deposit their energy. The absorbed dose would be expected to peak at the depth where the increase in the charged particle population and increase in dose is balanced by the decrease in dose associated with the attenuation of the photon beam. The maximum in the absorbed dose \mathbf{D} would occur likely at a depth \mathbf{x}_m somewhat less than the range of the most energetic electrons produced in the muscle phantom (i.e., $\leq 0.5 \text{ cm}$). At this depth of maximum dose, the dose \mathbf{D} will approximately equal the kerma \mathbf{K} . At greater depths, the absorbed dose exceeds the kerma because of the attenuation of primary photons and the build-up of secondary electrons with depth in the material where the forward direction of liberated electrons is favored because of momentum conservation considerations. Based upon the given photon μ value of **0.0626 cm⁻¹** for muscle and an electron, effective attenuation coefficient ν of **9.97 cm⁻¹** calculated from an empirical expression shown below (which

SOLUTIONS AND ANSWERS TO 1996 ABHP EXAM

normally would not be known by a candidate), the depth x_m in muscle where the maximum dose occurs can be approximated assuming a final transient equilibrium relationship between the kerma and dose:

$$* \quad x_m \approx \frac{1}{\nu - \mu} \ln\left(\frac{\nu}{\mu}\right) = \mathbf{0.512 \text{ cm}},$$

where for water equivalent material:

$$\nu \approx 13 (E_\gamma - 0.036)^{-1.37} = \mathbf{9.97 \text{ cm}^{-1}},$$

ν \equiv electron effective attenuation coefficient, $\text{cm}^{-1} = \mathbf{9.97 \text{ cm}^{-1}}$; and
 E_γ \equiv gamma photon energy, MeV = $\mathbf{1.25 \text{ MeV}}$ (not given in problem).

It also should be noted that the presence of bone at a depth of 1 cm will actually yield a dose enhancement as the muscle bone interface is approached, largely due to the increased backscatter of electrons in bone compared to muscle. This may lead to a dose maximum slightly beyond the interface.

Comment: As noted in the discussion above, a candidate would not be expected to calculate a specific depth where the dose is a maximum; yet, it would appear from the question that such a calculation may be required. Without more information, e.g., the photon energy and the back scattering from bone and muscle, neither a definitive numerical answer nor qualitative answer seems possible.

SOLUTIONS AND ANSWERS TO 1996 ABHP EXAM

QUESTION 5

GIVEN: Neutron activation is used to determine the neutron absorption cross sections of two materials in a sample, and a semilog plot of the assumed net counting rate versus time after irradiation is resolved into two components as shown in an attached figure:

m \equiv mass of sample = **100 g**;

ϕ \equiv uniform neutron fluence rate throughout sample = **$1.7 \times 10^{12} \text{ n cm}^{-2} \text{ s}^{-1}$** ;

F_1 \equiv fraction by weight of the target for the long lived component = **0.15**;

A_1 \equiv atomic weight of target for the long lived component = **14 g mole^{-1}** ;

F_2 \equiv fraction by weight of the target for the short lived component = **0.85**;

A_2 \equiv atomic weight of the target for the short lived component = **26 g mole^{-1}** ;

E_1 \equiv detection efficiency for long lived component = **0.45 c d^{-1}** ; and

Irradiation is long enough for saturation to be reached.

SOLUTIONS AND ANSWERS(*):

- * A. Explanation of how half lives of *two materials* (i.e. induced radionuclides) can be determined from the graph of counting rate versus time: The later counting rate data, assumed to represent only the response from the long lived component, is extrapolated as shown to zero time. These extrapolated counting rates for the long lived component are subtracted from the total net counting rate for both components to obtain the net counting rate for the short lived component, which is a straight line as shown in the semilog plot in the attached figure. The half-lives of each component can be obtained directly from the extrapolated lines for each component as the time required to reduce the initial net counting rate for each component (i.e., the intercept values) to $\frac{1}{2}$ these initial values.
- B. The initial activity $A(0)_L$ and half-life $T_{1/2}$ of the long lived component are estimated assuming:

$R(0)$ \equiv initial net counting rate from graph = $7.5 \times 10^7 \text{ cpm}$ = **$1.25 \times 10^6 \text{ cps}$** ; and so

* $T_{1/2}$ \equiv half-life read from graph at $3.75 \times 10^7 \text{ cpm}$ = **500 h**.

*
$$A_L(0) = \frac{R(0)}{E_1} = \frac{1.25 \times 10^6 \text{ cps}}{0.45 \text{ c d}^{-1}} = \mathbf{2.78 \times 10^6 \text{ Bq.}}$$

SOLUTIONS AND ANSWERS TO 1996 ABHP EXAM

- C. Assuming an initial rate $R(0)$ of 9×10^7 cpm or 1.5×10^6 cps, the microscopic absorption cross section σ (assumed here to also represent the activation cross section) is estimated from the saturation activity A_L for the long lived component, the calculated number of target atoms N_1 , and the given neutron fluence rate ϕ :

$$A_L = \frac{R(0)}{E} = \frac{1.5 \times 10^6 \text{ cps}}{0.45 \text{ c d}^{-1}} = 3.33 \times 10^6 \text{ Bq.}$$

The number N_1 of target atoms in the material being activated to the long lived component is calculated from the given data:

$$N_1 = \frac{m F_1}{A_1} (6.02 \times 10^{23} \text{ at mole}^{-1}) = 6.45 \times 10^{23} \text{ atoms.}$$

The activation cross section σ is then calculated:

$$* \quad \sigma = \frac{A_L}{\phi N_1} = \frac{(3.33 \times 10^6 \text{ s}^{-1})}{(1.7 \times 10^{12} \text{ cm}^{-2} \text{ s}^{-1})(6.45 \times 10^{23} \text{ at})} = 3.04 \times 10^{-30} \text{ cm}^2 \text{ at}^{-1}.$$

- * D. Five factors needed to be considered when evaluating materials as neutron shielding:
1. cost.
 2. weight.
 3. inclusion of nuclides that have high inelastic scattering cross sections to reduce energy of fast neutrons,
 4. inclusion of hydrogen which has a high elastic scattering cross section to reduce energy of fast and intermediate energy neutrons, and
 5. inclusion of a nuclide that has a high thermal neutron absorption reaction cross section and that does not lead to significant capture gamma radiation and induced radioactivity in the shield, e.g., boron whose main absorption reaction is $^{10}\text{B}(n,\alpha)^7\text{Li}$.

SOLUTIONS AND ANSWERS TO 1996 ABHP EXAM

QUESTION 6

GIVEN: ion exchange column used to process a liquid effluent stream containing a single radionuclide (See comments at end.):

- $T_{1/2}$ ≡ half-life of radionuclide = 15 days;
 - E_γ ≡ gamma photon energy = 1 MeV;
 - Y_γ ≡ yield of gamma photon = 1 γ d^{-1} ;
 - Γ ≡ gamma constant = **1.55 R h⁻¹ Ci⁻¹ m²**;
 - DF ≡ decontamination factor of column for radionuclide = 0.99;
 - D_c ≡ diameter of column = 0.5 m;
 - L** ≡ length of vertical column in height above floor = **9 m**;
 - S_L ≡ "concentration of radionuclide in the column", which, based on the given units, is considered to actually represent the activity per unit length = **0.015 Ci m⁻¹**;
 - ρ_{air} ≡ density of air = 1.293x10⁻³ g cm⁻³;
 - ρ_{Pb} ≡ density of lead = **11.36 g cm⁻³**; and
- a table of build-up factors **B** versus the number of relaxation lengths **z** and a table of μ/ρ and μ_{ab}/ρ values for air and lead for 1 MeV gamma photon.

SOLUTIONS AND ANSWERS(*):

A. Unshielded exposure rate $\dot{X}(0)$ at \perp distance **D** of **12 m** (assumed to be measured from the center of the column) and height **H** of **1 m** above floor based on the given assumptions and considering the column to represent a line source of length **L** of **9 m** having a linear specific activity S_L of **0.015 Ci m⁻¹**:

θ ≡ angle that line source subtends at exposure point and expressed in radians:

$$\theta = \tan^{-1}\left(\frac{H}{D}\right) + \tan^{-1}\left(\frac{L - H}{D}\right) = \mathbf{0.671 \text{ radian.}}$$

*
$$\dot{X}(0) = \frac{\Gamma S_L}{D} \theta = \mathbf{1.30 \times 10^{-3} R h^{-1}.}$$

SOLUTIONS AND ANSWERS TO 1996 ABHP EXAM

- B. Factor **F** by which unshielded exposure rate $\dot{X}(0)$ will be reduced by installation of a lead sleeve having a thickness **x** of **2 cm**:

Because the transmission through the shield of gamma photons emitted from different segments of the line source depends on the *crow-path* distance they travel through the shield toward the exposure point, the factor **F** only can be properly calculated by numerically integrating over the line source. This would be a tedious process unless a candidate were using a calculator such as a Hewlett Packard model 42S, which has numerical integration capability. Regardless, such a calculation is considered to be beyond the scope of this required question 6. An overestimate of the exposure rate transmission or value for the factor **F** is calculated for a \perp path through the shield corresponding to the shield thickness **x** of **2 cm** and number of relaxation lengths calculated:

$$\mu x = \left(\frac{\mu}{\rho} \right) \rho x = (0.0701 \text{ g cm}^{-2})(11.36 \text{ g cm}^{-3})(2 \text{ cm}) = \mathbf{1.59},$$

which yields a build-up factor **B** of **1.56** by linear interpolation between the given table values for 1 and 2 relaxation lengths. The transmission or factor **F** is then calculated:

*
$$F = B e^{-\mu x} = 1.56 e^{-1.59} = \mathbf{0.317}.$$

A candidate with an HP 42S calculator would first calculate the shielded exposure rate $\dot{X}(x)$ by integrating the following differential exposure rate $d\dot{X}$ over each differential segment dy of the line source:

$$\dot{X}(x) = \int_{y=0}^{y=L} d\dot{X} = \int_{y=0}^{y=L} \frac{\Gamma S_L dy}{(y - H)^2 + D^2} B e^{-\mu x \sec \phi} = \mathbf{3.87 \times 10^{-4} \text{ R h}^{-1}},$$

where:

- y** ≡ variable distance from the floor to the differential line segment dy , m;
- sec ϕ** ≡ $[(y - H)^2 + D^2]^{1/2} \div D$; and
- B** ≡ build-up factor calculated from a power function fit of (**B-1**) values versus the number of relaxation lengths **z**, which equals $\mu x \sec \phi$ and corresponds to the variable photon paths through the shield. The following fit equation is obtained from the given build-up factors by the *STATS* application on the HP 42S calculator:

$$(B - 1) = 0.369 z^{0.911}.$$

SOLUTIONS AND ANSWERS TO 1996 ABHP EXAM

The factor **F** or fraction of the exposure transmitted is calculated from the ratio of the value of $3.87 \times 10^{-4} \text{ R h}^{-1}$ for $\dot{X}(x)$ and the value of $1.30 \times 10^{-3} \text{ R h}^{-1}$ for $\dot{X}(0)$:

$$* \quad F = \frac{\dot{X}(x)}{\dot{X}(0)} = \frac{3.87 \times 10^{-4}}{1.30 \times 10^{-3}} = \mathbf{0.298},$$

which compares to the value of 0.317 overestimated from the simpler calculation shown above (See comments below.).

C. Reason why the equation $I = I_0 e^{-\mu x}$ does not apply to shielding of broad beams and thick shields:

* The dominant photon interaction in shields is Compton scattering, which leads to secondary or scattered photons that make the major contribution to the exposure rate on the outside of a thick shield. The given equation only accounts for the transmission and contribution of uncollided photons.

D. Shielding an uncompensated GM detector, which was previously calibrated with a precision source of the same radionuclide being processed, with lead to extend its range by a factor of 10 is expected to affect its response in comparison to the unshielded detector:

- * 1. The response for the given 1 MeV photons will decrease with lead shielding thickness because of attenuation of primary photons by photoelectric and Compton interactions. Electrons arising from these reactions, primarily in the lead and the wall of the detector, produce the response in the GM detector.
- * 2. For the same source strength, dead time losses are expected to be smaller in the shielded detector because of its lower response.
- * 3. The response to background radiation is expected to be lower in the shielded detector.

COMMENTS:

1. The given value of $1.55 \text{ R h}^{-1} \text{ Ci}^{-1} \text{ m}^2$ for the gamma constant is inconsistent with the given 1 MeV energy and 100% yield for the gamma ray, which could be used along with the reported energy absorption coefficient of 0.0279 cm g^{-1} to calculate $0.545 \text{ R h}^{-1} \text{ Ci}^{-1} \text{ m}^2$ for the gamma constant.
2. It is stated in the given information: "The concentration of the radionuclide in the column is 0.015 Ci/meter." The units given are not concentration units but rather units for linear specific activity, which are the units for a line source. The candidate has two choices: (1)

SOLUTIONS AND ANSWERS TO 1996 ABHP EXAM

assume a typographical error in the given units for the concentration and calculate the value of linear specific activity S_L as **0.00294 Ci m⁻¹** from the product of an assumed concentration **C** of **0.015 Ci m⁻³** and the cross sectional area **A** of **0.196 m²** calculated from $\pi D_c^2/4$ or (2) assume an error was made in the **concentration** description as shown in our solution.

3. The shielding calculation in part B directs the candidate: "Do not use rules of thumb." This could easily be interpreted by the more qualified candidate as requiring the more rigorous calculation shown above. The more rigorous solution demands a lot more time, especially if a more qualified candidate does not have the numerical integration capability and **STATS** application on the HP 42S calculator. Thus, this question is unfair to more qualified candidates especially if they might have resorted to a more rigorous manual integration over the line source. The ABHP directive should have stated: "Calculate the exposure rate transmission for normal incidence on a 2 cm thick lead shield to estimate the factor reduction in the unshielded exposure rate." The given build-up factors closely match the dose build-up factors for a point isotropic source in the Radiological Health Handbook. This geometry and the type of build-up factors (i.e., exposure) should have been stated in the problem.

SOLUTIONS AND ANSWERS TO 1996 ABHP EXAM

QUESTION 7

GIVEN: electron accelerator for which shielding is required:

- E** ≡ electron energy = **20 MeV**;
- I** ≡ peak current = **1,000 mA**;
- L** ≡ pulse length = **10⁻⁶ s**;
- F** ≡ pulse frequency = **10 s⁻¹**;
- Z_w** ≡ tungsten atomic number = **74**;
- Z_{Cu}** ≡ copper atomic number = **29**, and
five figures from NCRP 51.

SOLUTIONS AND ANSWERS (*):

- A. The concrete wall thickness (density = 2.35 g cm⁻³) to yield **0.5 mrem h⁻¹** when the dose point is at 90° from the beam direction and **5 meters** from the target is calculated:

The average beam current **I_{avg}** is calculated:

$$\mathbf{I_{avg} = (I)(L)(F) = (1000 \text{ mA})(10^{-6} \text{ s})(10 \text{ s}^{-1}) = 10^{-2} \text{ mA.}}$$

The output **$\dot{D}_{10} \text{ I}^{-1}$** of **2000 rads m² mA⁻¹ min⁻¹** at 90° for 20 MeV is read from Fig. E.1. The dose rate at **5 meters** in the 90° degree direction is then obtained:

$$\mathbf{D_{5m} = (I_{avg})(2000 \text{ rads m}^2 \text{ mA}^{-1} \text{ min}^{-1})(1/5 \text{ m})^2 = 0.800 \text{ rads/min,}}$$

which is equivalent to 0.800 rem/min or **800 mrem/min** since the quality factor for bremsstrahlung is unity.

The required transmission, **T**, is then calculated:

$$\mathbf{T = (0.5 \text{ mrem h}^{-1}) / ((800 \text{ mrem min}^{-1})(60 \text{ min h}^{-1})) = 1.04 \times 10^{-5}.$$

The bremsstrahlung at 90° has a lower effective energy than that in the beam direction, and Fig. E.6 may be used to incorporate the difference by finding an equivalent electron energy that would yield the desired photon energy distribution. From the figure, the equivalent electron energy is slightly above 10 MeV for the given 20 MeV incident electron energy. I shall assume 10 MeV. From Fig. E.8, using the **10 MeV curve**, we may read the required concrete thickness (density = 2.35 g cm⁻³) as **190 cm** for the required transmission value **T** of **1.04x10⁻⁵**:

* **required thickness = 190 cm.**

SOLUTIONS AND ANSWERS TO 1996 ABHP EXAM

- B. If the required transmission T is 10^{-4} and the existing concrete wall thickness x_1 is 2.5 feet or **30 inches**, the required additional lead thickness x_2 is calculated:

From the approach used in part A, a thickness x_1 of **30 inches** of concrete yields a transmission $T_1 = 10^{-2}$ for photons produced by **10 MeV** electrons. Thus, the additional transmission T_2 required of the lead is calculated:

$$T_2 = T/T_1 = 10^{-4}/10^{-2} = 10^{-2},$$

and this additional transmission may be obtained by adding **two tenth value layers** of lead. From Fig. E.14 (solid curve), at an electron energy of **10 MeV**, **1 TVL = 2.2 in.** Thus,

- * **The required added thickness is (2)(2.2 in) = 4.4 in = 11.2 cm of lead.**

- C. Relative neutron dose yields if copper is used instead of tungsten as the target and two reasons for the answer:

- * By neutron dose yield, I assume is meant the neutron dose at a fixed point away from the target per unit electron current on the target and that both targets are thick relative to the electron range. I would expect the neutron yield from the copper to be lower than that from the tungsten because (1) the binding energy per nucleon is higher in the lower Z copper than in the tungsten, and (2) the neutrons are produced by photoneutron interactions in the target; so the bremsstrahlung yield in the lower Z copper is less than that in the tungsten.

- * D. Four parameters that affect the reflection coefficient for scattered x-ray radiation are:

- (1) the energies of the incident photons,
- (2) the angle of incidence of the photons on the scatterer surface,
- (3) the angle of reflection from the scatterer surface, and
- (4) the material of construction of the scatterer.

SOLUTIONS AND ANSWERS TO 1996 ABHP EXAM

QUESTION 8

GIVEN: You are asked by a realtor for consultation on a problem with a radon measurement in a house where a sample in an unfinished basement next to an uncovered sump over a sampling time of 5 minutes yielded a reported radon concentration U of **19 pCi L⁻¹**, and you are given additional information:

- R** ≡ lifetime excess cancer mortality risk per WLM = **350 deaths/10⁶ person-WLM**;
L ≡ average life expectancy = **70 years**; and
F_{occ} ≡ residential occupancy factor = **0.5**.

SOLUTIONS AND ANSWERS(*):

- * A. The best definition for "working level" is "A. Any combination of the short lived radon daughters in 1 liter of air that results in the ultimate emission of 1.3×10^5 MeV of potential alpha energy."
- * B. Two justifications why I would not recommend remediation based upon the measurement include: 1. It is improper to compare the 5 minute measurement of 19 pCi L^{-1} to the EPA corrective action guide of 4 pCi L^{-1} , which is understood as an annual average concentration in living areas, and 2. the measurement in the basement next to an open sump could be many orders of magnitude greater than the actual annual average concentration in living areas.
- C. The lifetime risk of lung cancer death R_L is calculated based upon an equilibrium ratio F of **0.5**, a long term average radon concentration U of **19 pCi L⁻¹** (assumed in living areas), and other given information:

Comment: Candidates in this part are directed by the ABHP: "Assume that the radon progeny to radon activity ratio is 0.5." Does this mean that the total activity concentration of all short lived progeny is 50% of the given radon concentration U of **19 pCi L⁻¹**? Or, does it mean that the activity concentration of each progeny is 50% of this radon concentration? This second interpretation is not realistic for the following reason. When activity disequilibrium between radon and its progeny is present, the activity concentration of each progeny is less than that of its precursor. The first interpretation cannot be used to obtain an answer to this part because the actual concentration of each progeny is needed to calculate the number of working levels from all of the progeny. To obtain an answer to this part, the obtuse ABHP directive is interpreted as the equilibrium ratio F of 0.5. The equilibrium ratio F is rigorously defined:

$$F \equiv \frac{(\text{concentration } C \text{ of progeny in WL})(100 \text{ pCi L}^{-1} \text{ WL}^{-1})}{(\text{actual radon concentration } U \text{ in pCi L}^{-1})}$$

The above definition for F is based upon the fact that 100 pCi L^{-1} of ²²²Rn is required to

SOLUTIONS AND ANSWERS TO 1996 ABHP EXAM

support 1 WL of short lived progeny under secular equilibrium conditions when the concentration of each short lived progeny equals the radon concentration of 100 pCi L⁻¹. Thus, the numerator in the above equation represents the radon concentration only when secular equilibrium is present. When secular equilibrium is not achieved, removal of progeny from the air by mechanisms in addition to radioactive decay occurs, and the activity concentration of each progeny then is less than that of its precursor and the ²²²Rn. Under disequilibrium conditions, the actual concentration **U** of radon needed to support 1 WL of progeny exceeds 100 pCi L⁻¹. Thus, the effective secular equilibrium radon concentration calculated for the numerator in the above equation will be less than the actual radon concentration **U** shown in the denominator. The value of **F** has been found to vary from about 0.2 to 0.8, and the EPA has chosen an **F** value of 0.5 for the purpose of establishing its corrective action guide of 4 pCi L⁻¹ of radon, which can be more easily measured in homes than the individual concentrations of radon progeny. The concentration **C** of radon progeny in units of WL that corresponds to a radon concentration **U** of **4 pCi L⁻¹** (i.e., the EPA guide) and to an assumed equilibrium ratio **F** of **0.5** can be calculated by rearranging the equation above that defines the equilibrium ratio:

$$C = \frac{U F}{100 \text{ pCi L}^{-1} \text{ WL}^{-1}} = \frac{(4 \text{ pCi L}^{-1})(0.5)}{100 \text{ pCi L}^{-1} \text{ WL}^{-1}} = \mathbf{0.02 \text{ WL}},$$

which is the actual intent of the EPA guide for radon, i.e., to take corrective action when the long term average concentration **C** of radon progeny in living areas exceeds 0.02 WL.

Based upon the information and assumptions in this part and other given information, the lifetime risk **R_L** is calculated from exposure to **19 pCi L⁻¹** of radon for 70 years at 50% occupancy by first calculating the progeny concentration **C**:

$$C = \frac{U F}{100 \text{ pCi L}^{-1} \text{ WL}^{-1}} = \frac{(19 \text{ pCi L}^{-1})(0.5)}{100 \text{ pCi L}^{-1} \text{ WL}^{-1}} = \mathbf{0.095 \text{ WL}}.$$

The accumulative lifetime risk **R_L** is then calculated for the given occupancy factor **F_{occ}** of **0.5** where **170 h** is defined as **1 occupational month (M)**:

$$R_L = (0.5)(0.095 \text{ WL})\left(\frac{8760 \text{ h y}^{-1}}{170 \text{ h M}^{-1}}\right)(70 \text{ y})\left(\frac{350 \text{ deaths}}{10^6 \text{ person-WLM}}\right);$$

$$R_L = 0.060 \frac{\text{deaths}}{\text{person}} \text{ or } \frac{\mathbf{60 \text{ deaths}}}{\mathbf{1,000 \text{ persons}}}.$$

*

* D. Three potential sources of radon in a home and two remediation measures for each are:

1. Radon in soil gas from an open sump - remediations: fill and cap sump hole if sump is not needed or cap sump and connect pipe to cap and to fan on outside of house to

SOLUTIONS AND ANSWERS TO 1996 ABHP EXAM

maintain a negative pressure in sump and under basement floor.

2. In an unfinished basement, radon in soil gas enters through various pathways, e.g. through cracks where floor abuts foundation, etc. - remediations: seal all pathways with plastic based sealant and if this does not work put in sub-floor suction system similar to that in 1 above.
3. Radon in well water escapes into air when water is used in a home, e.g., from showers, etc. - remediations: place aeration system on water intake line and strip radon from water before it enters home or place activated charcoal system on water intake line to adsorb radon from the water.

* E. Answers to two questions regarding "unattached fraction":

1. The "unattached fraction" represents the fraction of any given radon progeny unattached to condensation nuclei in the air and existing as free atoms or ions, which have a much higher diffusion coefficient than the attached species.
2. The increase shown in the figure of the tracheobronchial dose per unit exposure to radon progeny in rad/WLM as the unattached fraction increases is due to the fact that unattached radon progeny have a much higher diffusion coefficient than the attached species and these free atoms and ions quickly diffuse and deposit on walls of the tracheobronchi where they decay and deliver most of the dose. The activity in the inhaled air that deposits by diffusion deposition on the bronchial epithelium is essentially independent of the breathing rate; therefore, for a fixed unattached fraction the total dose is directly proportional to the total radon progeny exposure in WLM, a unit applied to occupational exposure in a mine.

* F. Two factors (cautions) that should be considered in applying mining data to residential homes in the estimation of risks to occupants include:

1. Miners were heavy smokers and their epidemiological data will most likely overestimate the risk to non-smoking occupants of a home especially if radon and smoke acted synergistically in causing the high incidence of lung cancers observed in miners.
2. Miners also may have been exposed to other carcinogens, e.g., diesel smoke, dusts including radioactive species such as uranium, thorium, and their progeny that are not present in a home but which might have been partially responsible for lung cancers observed in miners.

* G. The statement best representing the action recommended by the EPA when the average, long-term radon concentration in living areas is determined to be 10 pCi L^{-1} is "C. Fix your home; take action to remediate the radon problem."

SOLUTIONS AND ANSWERS TO 1996 ABHP EXAM

QUESTION 9

GIVEN: nuclear power plant at which diver is to repair floor plates in spent fuel pool:

t	≡	estimated job duration = 20 minutes = (1/3) h.;
\dot{H}_{spot}	≡	gamma dose equivalent rate at 1 foot from spot = 600 rem h⁻¹ ;
d_p	≡	distance from hot spot to floor plates = 3 feet = 91.4 cm ;
x	≡	distance from point of γ dose rate measurement to plates = 2 feet = 61.0 cm ;
TVL	≡	tenth value layer for water = 60 cm ;
H_{acc}	≡	diver dose of record for current year = 2,500 mrem ;
E_g	≡	average gamma energy = 1 MeV;
E_o	≡	maximum beta energy from tritium = 0.0186 MeV;
E_{avg}	≡	average beta energy from tritium = 0.0057 MeV ;
T_r	≡	radiological half-life of tritium = 4,490 days ;
T_b	≡	biological half life of tritium = 10 days ;
V	≡	volume of water in body of RM = 42,000 mL (actual value is 43 L); and
m	≡	mass of soft tissue in body = 60,000 g (actual value is 63 kg).

SOLUTIONS AND ANSWERS (*):

A. The dose equivalent rate at the floor plates is calculated:

I assume the straight line distance from the hot spot to the plates is 3 feet, that gamma radiation from the hot spot penetrates only water and is the only source of radiation at the plates, and that the hot spot behaves as a point isotropic source. It should be noted that the distance of 3 feet is measured along the pool bottom and the dose point of interest is on the surface of the plates; the effect of scattered radiation from the pool bottom may be somewhat different from that in water; and the TVL of 60 cm may be in some error, but this will be neglected. The dose equivalent rate at the plates is estimated:

$$* \quad \dot{H}_{\text{plates}} = (\dot{H}_{\text{spot}})(1 \text{ ft}/3 \text{ ft})^2(1/10)^{61.0/60} = \mathbf{6.42 \text{ rem h}^{-1}}.$$

B. The maximum stay time **T** is calculated for the diver to remove the plates (no planned special exposure allowed), if the dose equivalent rate \dot{H} at the floor plates is **5,000 mrem h⁻¹**:

The solution is based on the current 10CFR20 limit of 5 rem in the current year. Because the worker has already received an accumulated dose H_{acc} of **2,500 mrem** for the year, he would be allowed a maximum dose H_L of **2,500 mrem** more from this job; so **T** is calculated:

$$* \quad T = (H_L)/(\dot{H}) = 2500 \text{ mrem}/5000 \text{ mrem h}^{-1} = \mathbf{0.500 \text{ h}}.$$

SOLUTIONS AND ANSWERS TO 1996 ABHP EXAM

* C. Five controls which could be applied to ensure the diver remains below regulatory limits are summarized:

- (1) control job duration;
- (2) monitor dose rate in the vicinity of the diver with an underwater detector;
- (3) have the diver maintain a maximum reasonable distance from the hot spot (e.g., work from the side away from the spot);
- (4) provide localized shielding to reduce dose rates (i.e., cover the hot spot with lead or steel), and
- (5) provide direct instruction/advice to the diver through underwater communication.

D. Solutions and answers to the three questions in this part are:

- * (1) Five actions that I would take upon discovering contamination on the diver's upper legs from a leak in his suit after the dive include:
- (a) remove remainder of suit and check for additional contamination;
 - (b) attempt to get a reasonably quantitative measure of the skin contamination with an appropriate instrument (e.g., thin window G-M detector) and attempt to sample the radioactivity with a gentle wipe or with sticky tape (on dry skin);
 - (c) decontaminate the worker;
 - (d) from the wipe in (b) attempt to identify the radionuclide(s);
 - (e) from the information obtained in (d) and from the instrument measurements in (b), calculate the dose rate to the live skin (normal calculation is averaged over a 1 cm^2 area at a depth of 7 mg cm^{-2}).
- * (2) The diver voided his bladder after the dive, and two hours later a urine sample was collected and yielded a measured tritium concentration U of $0.01 \text{ } \mu\text{Ci mL}^{-1}$. Answer for whether or not the measurement was valid for dosimetric purposes:

Presumably the tritium uptake occurred via skin penetration, likely when tritiated water contacted the skin. The exchange of tritiated water with water in the skin is rapid and the absorbed tritium quickly distributes in the systemic circulation and distributes uniformly among all the body water, including water that goes to waste via urine. As such, even though two hours is a relatively short time, I would expect that the urine sample was reasonably representative of the concentration of tritium in the body water if the entire bladder contents was voided prior to obtaining the sample that was analyzed.

SOLUTIONS AND ANSWERS TO 1996 ABHP EXAM

- (3) The diver's committed effective dose equivalent (CEDE), assuming no intervention to reduce the tritium concentration of part D.2 is calculated:

The CEDE may be calculated by assuming that tritium is distributed in body water uniformly among all the soft tissues of the body. The committed effective dose equivalent, in rem, is calculated for an activity concentration in soft tissue, A/m in units of $\mu\text{Ci g}^{-1}$, from the following equation when given quantities in the equation have the bolded units shown here and in the general given information:

$$* \quad CEDE = 51.1 \frac{A E_{avg}}{m \lambda_e} = (51.1) \frac{(420)(0.0057)}{(60,000)(0.0695)} = \mathbf{0.0293 \text{ rem}},$$

where:

$$\begin{aligned} T_{\text{eff}} &\equiv T_r T_b / (T_r + T_b) = \mathbf{9.98 \text{ days}}; \text{ so} \\ \lambda_e &\equiv \text{effective removal rate constant} = (\ln 2) / T_{\text{eff}} = \mathbf{0.0695 \text{ day}^{-1}}; \\ A &\equiv U V = (0.01 \mu\text{Ci/mL})(42,000 \text{ mL}) = \mathbf{420 \mu\text{Ci}}, \text{ which is assumed to be the} \\ &\quad \text{initial activity in the body.} \end{aligned}$$

SOLUTIONS AND ANSWERS TO 1996 ABHP EXAM

QUESTION 10

GIVEN: a plutonium fire in a glove box, whose entire activity is released as smoke at a constant rate over a 20 minute period into a room, which has its emergency ventilation system automatically activated:

- h** ≡ stack height = **10 m**;
- T** ≡ burn and release time = 20 minutes = **1,200 s**;
- m** ≡ mass of ²³⁹Pu in glove box = **500 g**;
- V** ≡ room volume = **108 m³**;
- T_{1/2}** ≡ half-life of ²³⁹Pu = **24,100 y**; so
- λ** ≡ decay constant of ²³⁹Pu = **9.12x10⁻¹³ s⁻¹**;
- F** ≡ emergency ventilation of room = 7 m³ min⁻¹ = **0.117 m³ s⁻¹**; so
- K** ≡ ventilation removal rate constant = F/V = **1.08x10⁻³ s⁻¹**; and
- k** ≡ total removal rate constant K + λ ≈ K = **1.08x10⁻³ s⁻¹**;
- P** ≡ maximum HEPA fractional penetration = **0.0005**;
- DAC** ≡ derived air concentration for ²³⁹Pu = **2x10⁻¹² Ci m⁻³**;
- ū** ≡ mean wind speed = **7 m s⁻¹**;

Stability class **C**;

Graphs of σ_y and σ_z in units of **meters**; and

Equation for downwind concentration χ in units of **Ci m⁻³** when parameters have units shown and release rate **Q'** has units of **Ci s⁻¹**.

SOLUTIONS AND ANSWERS(*):

A. The total activity **A** of ²³⁹Pu in the glove box is calculated:

$$* \quad A = (9.12 \times 10^{-13} \text{ s}^{-1}) \left(\frac{500 \text{ g}}{239 \text{ g mole}^{-1}} \right) \left(\frac{6.02 \times 10^{23} \text{ at mole}^{-1}}{3.7 \times 10^{10} \text{ at s}^{-1} \text{ Ci}^{-1}} \right) = \mathbf{31.0 \text{ Ci}}$$

B. The activity concentration **C(T)** is estimated in the room **1,200 s** after the start of the fire assuming an entire activity **A** of **50 Ci** is released at a constant rate over **T** of **1,200 s**:

$$* \quad C(T) = \frac{A}{V} \left(\frac{1 - e^{-kT}}{kT} \right) = \mathbf{0.259 \text{ Ci m}^{-3}}$$

SOLUTIONS AND ANSWERS TO 1996 ABHP EXAM

- C. The time **t** is calculated for the concentration **C(t)** to equal the product of the **DAC** of **2x10⁻¹² Ci m⁻³** and the protection factor **PF** of **10,000** for a pressure demand SCBA beginning at an initial concentration **C(0)** of **3x10⁻⁴ Ci m⁻³** and for the removal rate constant **k** of **1.08x10⁻³ s⁻¹**:

$$C(t) = (\text{DAC})(\text{PF}) = 2 \times 10^{-8} \text{ Ci m}^{-3}, \text{ and thus}$$

$$* \quad t = \frac{1}{k} \ln \left(\frac{C(0)}{C(t)} \right) = 8,900 \text{ s} = 2.47 \text{ h.}$$

- D. The downwind ground concentration **χ** on the plume centerline is calculated from the given equation, stability class C, and other given information shown in bolded units:

$$Q' = \text{stack release rate} = (3 \times 10^{-4} \text{ Ci m}^{-3})(0.117 \text{ m}^3 \text{ s}^{-1})(0.0005) = 1.76 \times 10^{-8} \text{ Ci s}^{-1};$$

$$\sigma_y = \text{cross wind dispersion coefficient at 1,000 m from graph for stability class C} = 110 \text{ m}; \text{ and}$$

$$\sigma_z = \text{vertical dispersion coefficient at 1,000 m from graph for stability class C} = 67 \text{ m}; \text{ and}$$

$$y = z = 0; \text{ so the given equation reduces and yields answer:}$$

$$* \quad \chi = \frac{Q'}{\pi \sigma_y \sigma_z \bar{u}} e^{-\frac{1}{2} \frac{h^2}{\sigma_z^2}} = 1.07 \times 10^{-14} \text{ Ci m}^{-3}.$$

SOLUTIONS AND ANSWERS TO 1996 ABHP EXAM

QUESTION 11

GIVEN: A patient is given ^{131}I therapy:

I \equiv oral dose = **$1 \times 10^5 \mu\text{Ci}$** ;

F \equiv fraction of oral dose assumed to be taken up instantaneously into thyroid = **0.3**;

T_r \equiv radiological half-life of ^{131}I = **8.05 days**;

T_b \equiv biological half-life in thyroid = **90 days**; so

k \equiv total removal rate constant = $(\ln 2)/T_r + (\ln 2)/T_b =$ **0.0938 day^{-1}** ;

m \equiv mass of thyroid = **20 g**;

ϵ \equiv average beta energy = **0.19 MeV**; and

1 eV = 1.6×10^{-19} j.

SOLUTIONS AND ANSWERS(*):

A. The absorbed dose **D** to the patient's thyroid over the first year is calculated:

Comment: The dose **D** is essentially the ultimate dose calculated, and this dose would normally include a significant component from gamma radiation, which cannot be calculated because sufficient information is not given in the question. The dose **D** in the unit **Gy** is obtained from given quantities in the bolded units shown above:

$$* \quad D = 0.511 \frac{F I \epsilon}{m k} = \frac{(0.511)(0.3)(1 \times 10^5)(0.19)}{(20)(0.0938)} = \mathbf{1,550 \text{ Gy.}}$$

B. Answers to two questions regarding patient room preparation:

- * 1. Four radiation protection concerns include: (1) external radiation received by staff, (2) external radiation received by visitors, (3) internal radiation received by family or friends, and (4) spread of contamination.
- * 2. Four specific measures taken in room preparation include: (1) posting sign on door of room that informs staff and visitors that room contains a patient being treated by radiation therapy, (2) providing properly labeled vessels with instructions for the collection and processing of all excreta from patient, (3) providing properly labeled hamper with instructions for the collection and processing of all bedding and clothing worn by patient, and (4) installing yellow rope barriers at a specified distance from patient with signs providing appropriate instructions to visitors and staff.

SOLUTIONS AND ANSWERS TO 1996 ABHP EXAM

- * C. Four radiation protection measures/controls that should be implemented for the protection of the hospital staff include: 1. providing training of staff in the proper care of a radiation therapy patient, 2. informing patient of procedures to be followed to minimize spread of contamination and external radiation received by all persons whom they may contact, 3. removal of contaminated bedding and clothing, and 4. removal of excreta and flushing it through toilet, especially urine, as soon as it is collected or have patient go directly to toilet and flush all excreta if patient is ambulatory.

- * D. Two radiation **protection** concerns regarding patient's resumption of activities as a radiation worker at a nuclear power facility include external radiation exposure and internal radiation exposure received by persons having contact with the patient during such activities if patient were allowed to leave hospital with sufficient activity to warrant such concern.

Comment: However, of greater concern is the disruption of the routine radiation protection activities that could occur at the plant when other uninformed staff might have to respond to portal alarms, contamination checks of this individual, or reports of significant ¹³¹I found in samples, e.g., air filter samples or smears that the individual might have handled as a health physics technician.

SOLUTIONS AND ANSWERS TO 1996 ABHP EXAM

QUESTION 12

GIVEN: diagnostic medical facility concerned with breast dose reduction and information for an AP chest x-ray examination:

- H_s \equiv maximum dose equivalent in tissue at midfield position of skin entrance = **300 μSv** ;
- f \equiv exposure to dose conversion factor = **$0.927 \text{ cGy}/2.58 \times 10^{-4} \text{ C kg}^{-1}$** ;
- w_T \equiv tissue weighting factor (table of tissue weighting factors given), and
- F_T \equiv fractional mean organ dose equivalent from maximum entrance dose equivalent (table given in units of **$\mu\text{Sv}/\mu\text{Sv}$** entrance dose).

SOLUTIONS AND ANSWERS(*):

A. The effective dose equivalent, H_E , for an AP chest x-ray is obtained from the equation:

$$H_E = \sum w_T H_T = H_s \sum w_T F_T ,$$

where H_T is the dose equivalent delivered to tissue T. Using the values of H_s , w_T , and F_T given in the problem, we obtain:

* $H_E = 300 \mu\text{Sv} [(0.15)(0.75)+(0.12)(0.1)+(0.12)(0.4)+(0.03)(0.2)+(0.03)(0.15)+(5)(0.06)(0.4)+(0.06)(0.2)];$
 $= 94.5 \mu\text{Sv}.$

Comment: The sum of the weighting factors in the above calculation exceeds unity, which does not make any practical sense. However, the recommendations embodied in ICRP Publication 26 do not clarify how this discrepancy should be handled, especially for external radiation. The concept of effective dose equivalent was primarily introduced for internal radiation, but this question relates to external radiation.

- * B. A simple modification to the chest x-ray examination to greatly reduce the mean organ dose to the breast compared to taking an AP chest x-ray is to perform a posterior-anterior exposure (PA) rather than an AP exposure.
- * C. Two practical disadvantages in using H_E (rather than skin entrance exposure) are:
 - (1) The tissue dose distribution can differ significantly among patients of different body sizes and shapes, and the estimate of patient dose may be in significant error when reference man tissue weighting factors are used.
 - (2) The tissue dose distribution changes as effective photon energy changes, and depending on the technique used by a particular technologist for a particular patient, the risk estimate from the estimated H_E value may be different, for a given entrance exposure, than that when a different technique (e.g., kvp) is used by the same or a different

SOLUTIONS AND ANSWERS TO 1996 ABHP EXAM

technologist.

- * D. It is not appropriate to use the concept of H_E in radiotherapy risk assessment and communication because in radiotherapy acute high doses are typically delivered to selected volumes of tissue in order to kill malignant cells. Often a single organ/tissue is involved and the radiation benefit is considered to far outweigh the risk to that tissue (or other tissues that may be incidentally irradiated at lower doses) of inducement of cancer or genetic effects. Indeed, the risk factors that apply to a healthy tissue in a healthy individual may be quite different from the risk factor that applies to a diseased tissue in an ill individual. The concept of H_E and associated risk is intended for application in cases of chronic, low-level exposure (actually of populations, not individuals) and is not meant to be used to interpret risk associated with acute, large doses of radiation to an individual.
- * E. Three factors that the exposure to glandular dose conversion factor depend upon are:
- (1) the energy quality of the incident x-rays,
 - (2) the size and shape of the irradiated breast, and
 - (3) the density of the breast tissue.
- F. If the average glandular dose \bar{D}_g to an average patient is **1.07 mGy** and the (average) incident exposure in air needed to produce a proper density image is **$1.8 \times 10^{-4} \text{ C kg}^{-1}$** , then the average glandular dose conversion factor \bar{D}_{gN} in **$\text{mGy}/2.58 \times 10^{-4} \text{ C kg}^{-1}$** is calculated:
- I shall assume that the figure $1.8 \times 10^{-4} \text{ C kg}^{-1}$ given for the needed exposure represents the average incident exposure to an average patient. The value of \bar{D}_{gN} then would be obtained in the specified units:
- *
$$\bar{D}_{gN} = (1.07 \text{ mGy}/1.8 \times 10^{-4} \text{ C kg}^{-1})(2.58 \times 10^{-4} \text{ C kg}^{-1}/2.58 \times 10^{-4} \text{ C kg}^{-1}) = \mathbf{1.53 \text{ mGy}/2.58 \times 10^{-4} \text{ C kg}^{-1}}$$
- * G. Five quality assurance tests that a medical physicist would perform on mammographic x-ray equipment are:
- (1) accuracy of kvp,
 - (2) accuracy of timer,
 - (3) half value layer measurement,
 - (4) linearity and reproducibility of mA, and
 - (5) measurement of representative entrance surface exposure.

SOLUTIONS AND ANSWERS TO 1996 ABHP EXAM

QUESTION 13

GIVEN: You, the Radiation Safety Officer, are expected to be the expert about the biological effects and exposure criteria for RF and ELF electromagnetic radiation.

SOLUTIONS AND ANSWERS(*):

* A. Definitions are given:

1. Electric Field Strength is the vector force per unit charge on a point charge at a point in space. It is usually symbolized by **E** and has units of volts/meter.
2. Magnetic Field Strength, usually symbolized by **H** and having the units of amperes/meter, is also a vector quantity and represents the quotient of the magnetic flux density, symbolized by **B**, by the permeability of the medium, symbolized by μ .
3. The Poynting Vector is the vector (cross) product of the electric and magnetic field strengths, and it is given by

$$\mathbf{E} \times \mathbf{H} = E H \sin \theta ,$$

where θ is the angle between the electric and magnetic field strength vectors, **E** is the magnitude of **E**, and **H** is the magnitude of **H**. The cross product, also a vector quantity, is usually symbolized by **S** and given in units of watts per square meter when **E** and **H** are defined in the units above.

* B. An example of ELF electromagnetic radiation is the 50-60 Hz radiation resulting from the generation, propagation, and use of common household, commercial and industrial alternating current.

SOLUTIONS AND ANSWERS TO 1996 ABHP EXAM

- * C. Following is a sketch of a plane, sinusoidal electromagnetic wave showing two oscillating **E** and **H** fields:

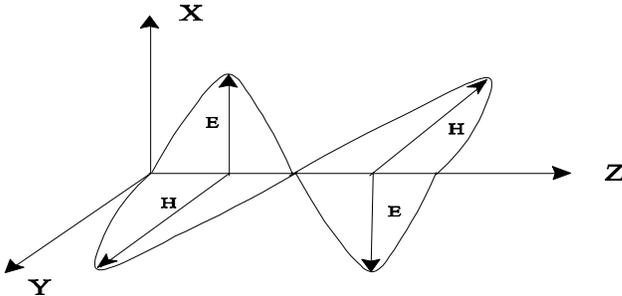


Figure 13.1. Schematic of Electromagnetic Wave.

The electric and magnetic field vectors, **E** and **H**, shown in the sketch lie in the X-Z and Y-Z planes, respectively, and are perpendicular to each other as well as to the Z-direction of propagation.

- * D. Two sources of electromagnetic radiation in VDTs are (1) the flyback transformer, and (2) the electron deflection coils. Both of the above are part of the horizontal sweep circuitry. The electromagnetic fields from these are pulsed and produce a wide range of harmonic frequencies. Electric field strengths at operator locations are typically about 50 V/m and magnetic field strengths about 0.50 A/m. Very little energy comes from frequencies greater than 200 kHz.
- * E. The biological effect of RF electromagnetic radiation that is the primary basis for establishing RF exposure criteria is the thermal effect.

SOLUTIONS AND ANSWERS TO 1996 ABHP EXAM

QUESTION 14

GIVEN: You are asked to estimate the dose to the tissue of an animal that was injected with a beta emitting radioisotope of phosphorous having a beta yield of 100%. Additional information includes the specified maximum energy E_{\max} in MeV, specified average energy E_{avg} in **MeV**, radiological half-life T_r in **days**, approximate absorption half-value layer $X_{1/2}$ in **mg cm⁻²**, equation for beta particle range **R** in **mg cm⁻²**, equation for fraction **f** of beta energy converted to photons, biological half-life T_b of **19 days** in soft tissue, atomic numbers **Z**, and for air and photons with energy **E** of **0.7 MeV**: the mass attenuation coefficient μ/ρ of $7.56 \text{ m}^2 \text{ kg}^{-1}$ and mass energy absorption coefficient μ_{en}/ρ of **2.93 m² kg⁻¹**.

Comment: The given equation shown below for the beta particle range **R** was incorrectly reproduced from the expression given in the Radiological Health Handbook. The constant 0.954 shown in the expression for the exponent of E should have been 0.0954. The values given for the mass attenuation coefficient μ/ρ and mass energy absorption coefficient μ_{en}/ρ are 1,000 times the actual values. These incorrect data are used in the solutions shown below. Answers calculated from the correct data and equation for the range are shown after the incorrect answers.

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A. The dose **D** in **Gy** to an organ of a live animal is calculated for an initial concentration **C(0)** of **1.35 $\mu\text{Ci g}^{-1}$** ($5 \times 10^7 \text{ Bq kg}^{-1}$) over a period **T** of **10 days** for ³²P (E_{avg} of **0.695 MeV** and T_r of **14.3 days**) from the following equation using the values for the symbols and bolded units shown above and for **k**:

I assume the given biological half-life T_b of **19 days** applies and that the average beta energy is completely absorbed in the organ.

$$k \equiv \text{total removal rate constant} = (\ln 2)/T_r + (\ln 2)/T_b = \mathbf{0.0850 \text{ day}^{-1}}.$$

$$* \quad D = 0.511 \frac{C(0) E_{\text{avg}}}{k} (1 - e^{-k T}) = \mathbf{3.23 \text{ Gy}}.$$

SOLUTIONS AND ANSWERS TO 1996 ABHP EXAM

- B. The shallow doses **D** to the fingers of a person wearing gloves having a mass density thickness of **10 mg cm⁻²** is calculated for either ³²P or ³³P over a time **T** of **(1/6) h** for a person holding an organ containing either radioisotope with an associated energy spatial equilibrium beta dose rate **\dot{D}** of **0.5 Gy h⁻¹** to the organ:

I assume in either case 1 or 2 below that radioactive decay can be neglected, that the given half-value layer **X_{1/2}** applies to the attenuation of the respective ³²P or ³³P beta particles through a total thickness **X** of **17 mg cm⁻²** (gloves + 7 mg cm⁻² of skin) provided this thickness does not exceed the range of beta particles, and that the dose rate at the surface of the gloves is 1/2 of the energy spatial equilibrium dose rate **\dot{D}** to the organ. In either case, the shallow dose **D** is then calculated by the following equation using the values for the symbols and bolded units shown here and below:

$$D = \frac{1}{2} \dot{D} T e^{-\left(\frac{\ln 2}{X_{1/2}}\right)X}$$

1. The shallow dose **D** for ³²P is calculated for the given **X_{1/2}** of **150 mg cm⁻²**:

*
$$D = \frac{1}{2} (0.5 \text{ Gy h}^{-1}) \left(\frac{1}{6} \text{ h} \right) e^{-\left(\frac{\ln 2}{150}\right)(17)} = \mathbf{0.0385 \text{ Gy}},$$

where the range **R** calculated from the given equation is **617 mg cm⁻²**, which exceeds the absorber thickness **X** of **17 mg⁻²**:

$$R = 412 (E)^{(1.265 - 0.954 \ln(E))} = 412 (1.71)^{(1.265 - 0.954 \ln(1.71))} = \mathbf{617 \text{ mg cm}^{-2}},$$

which is calculated as **790 mg cm⁻²** from the correct equation.

2. The shallow dose **D** for ³³P is calculated for the given **X_{1/2}** of **6 mg cm⁻²**:

*
$$D = \mathbf{0},$$

because the range **R** calculated from the given equation is **11.2 mg cm⁻²**, which is less than the absorber thickness **X** of **17 mg⁻²**:

$$R = 412 (0.249)^{(1.265 - 0.954 \ln(0.249))} = \mathbf{11.2 \text{ mg cm}^{-2}}.$$

For the correct data and range equation, the range **R** is calculated as **59.0 mg cm⁻²** and the

SOLUTIONS AND ANSWERS TO 1996 ABHP EXAM

dose is calculated:

$$* \quad D = \frac{1}{2} (0.5 \text{ Gy h}^{-1}) \left(\frac{1}{6} \text{ h} \right) e^{-\left(\frac{\ln 2}{6}\right)(17)} = \mathbf{0.00585 \text{ Gy}},$$

- * C. Attenuation of ^{32}P beta particles is approximately exponential over thicknesses less than the range because the beta particles are emitted with energies from zero up to the maximum energy.
- D. The bremsstrahlung dose rate \dot{D} in air is calculated at a distance d of **0.30 m** from a mass m of **0.050 kg** of an organ (Z of **7.1**) having a concentration C of **$5 \times 10^7 \text{ Bq kg}^{-1}$** of ^{32}P assuming that the energy E of each photon is the average beta energy E_{avg} of **0.7 MeV** for which air has the given incorrect μ_{en}/ρ value of **$2.93 \text{ m}^2 \text{ kg}^{-1}$** :

The fraction f of beta energy converted to photons is calculated from the given equation:

$$f = 3.5 \times 10^{-4} Z E = 3.5 \times 10^{-4} (7.1)(0.7) = \mathbf{1.74 \times 10^{-3}}.$$

The dose rate \dot{D} in air is then calculated neglecting the very slight photon attenuation in air and in the tissue mass:

$$\begin{aligned} \dot{D} &= \frac{m C E f}{4 \pi d^2} \left(\frac{\mu_{\text{en}}}{\rho} \right) (1.6 \times 10^{-13} \text{ J MeV}^{-1}) \left(\frac{1 \text{ Gy}}{\text{J kg}^{-1}} \right) \left(\frac{3,600 \text{ s}}{\text{h}} \right); \\ \dot{D} &= \frac{(0.05 \text{ kg})(5 \times 10^7 \text{ s}^{-1} \text{ kg}^{-1})(0.7 \text{ MeV})(1.74 \times 10^{-3})(2.93 \text{ m}^2 \text{ kg}^{-1})}{4 \pi (0.3 \text{ m})^2} \\ & \quad (1.6 \times 10^{-13} \text{ J MeV}^{-1}) \left(\frac{1 \text{ Gy}}{\text{J kg}^{-1}} \right) \left(\frac{3,600 \text{ s}}{\text{h}} \right) = \mathbf{4.54 \times 10^{-6} \text{ Gy h}^{-1}}, \end{aligned}$$

*

which is calculated as **$4.54 \times 10^{-9} \text{ Gy h}^{-1}$** from the correct value of **$0.00293 \text{ m}^2 \text{ kg}^{-1}$** for μ_{en}/ρ .