SOLUTIONS AND ANSWERS TO 1998 ABHP EXAM

QUESTION 1

GIVEN: Tritium production facility and given or calculated quantities:

ALI ≡ annual limit on intake = **80,000 µCi;**
λ ≡ decay constant = (ln2)/(12.3)(365 days) = **$1.54x10^{-4}$ day^{-1}**;
V ≡ volume of water in whole body = **43 L**;
m ≡ mass of soft tissue in body = 65,000 g (Actual value is 63,000 g.);
F ≡ daily water loss from body of RM = **3 L day^{-1}**; so
K ≡ biological removal rate constant = F/V = **$6.98x10^{-2}$ day^{-1}**; so
k ≡ total removal rate constant = λ + K = **$6.99x10^{-2}$ day^{-1}**;
F_u ≡ daily volume of urine for RM = 1.4 L day^{-1}.

SOLUTIONS AND ANSWERS(•):

• A. Two airborne tritium monitoring techniques and their advantages and disadvantages include: (1) flow through Kanne ionization chamber: **advantage**: provides essentially an instantaneous measurement of the total airborne concentration of HT0 + T_2 and **disadvantage**: does not distinguish between T_2 and HTO, which has a much greater dose significance than T_2 and (2) measurement of tritium in the form of tritiated water vapor (HTO) in a measured mass of a condensed water vapor sample with LSC to obtain its specific activity S_A in µCi g^{-1} and the measurement of the relative humidity and temperature at the time of sampling to obtain an estimate of the absolute airborne concentration C_W in g cm^{-3} of water vapor; the HTO airborne concentration C_{HTO} in units of µCi cm^{-3} is then obtained from the product of S_A and C_W : **advantage**: provides an accurate measurement of the HTO airborne concentration, which contributes most of the dose when mixtures of T_2 and HTO are present in the air and **disadvantage**: requires sampling and analysis, which does not provide continuous monitoring of the airborne tritium.

B. The uptake **U** is calculated from the measured urine concentration **C_u(t)** of 500 dpm mL^{-1} or **500,000 dpm L^{-1}** observed at a time **t** of **60 days** after the uptake:

$$\langle C_u(t) \rangle \;=\; \frac{U}{V}\; e^{-kt};\; so$$

• $$\boldsymbol{U} \;=\; C_u(t)\; V\; e^{kt} \;=\; \boldsymbol{1.43x10^{9}\;\; dpm}.$$

C. The target organ is specified, and the total effective dose equivalent, **TEDE**, is calculated for an intake **I** of **1,600 µCi** and external dose equivalent **H_{ext}** of **100 mrem**:

• $$\boldsymbol{TEDE} \;=\; I \left(\frac{5{,}000\;\; mrem}{ALI} \right) \;+\; H_{ext} \;=\; \boldsymbol{200\;\; mrem\;\; for\;\; WB\;\; as\;\; target\;\; organ}.$$

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QUESTION 2

GIVEN: An ion chamber vented to the atmosphere:

V = **200 cm^3**; Bias Voltage = 25 V; and conversion constants.

SOLUTIONS AND ANSWERS(•):

A. Ion current **I** in amperes is calculated for an exposure rate $\dot{\mathbf{X}}$ of **0.1 R h^{-1}** at STP conditions:

$$I \frac{(1C\ s^{-1}A^{-1})(3{,}600s\ h^{-1})}{(200\ cm^3)(1.29x10^{-6}\ kg\ cm^{-3})(2.58x10^{-4}\ C\ kg^{-1}\ R^{-1})} = \dot{X} = 0.1\ R\ h^{-1};\ so$$

• $$\boldsymbol{I = 1.85x10^{-12}\ A.}$$

• B. The exposure rate reading of the meter for the ion chamber will be:

1. less than the actual exposure rate in Santa Fe (elevation 6,375 ft.) when calibrated in San Diego (elevation 120 ft.) because air density will be less than when calibrated.
3. more than actual exposure rate at 0 degrees C when calibrated at 25 degrees C because air density will be greater than when calibrated.

• C. If the ion chamber instrument is calibrated for absorbed dose rate to air in rad h^{-1} for photons, the dose rate reading on the meter would have to be multiplied by a correction factor to obtain the gamma skin dose rate. For most photons, ($E_\gamma \geq 50$ keV) the correction factor would be the tissue to air mass collision stopping power ratio for the electron distribution produced by photon interactions and present at a tissue depth of 7 mg cm^{-2}. For very low energy photons, which require little penetration depth to yield an equilibrium distribution of secondary electrons, the correction factor would be the tissue to air mass energy absorption coefficient ratio for the photons. Both of these ratios are about 1.1; so the reading would be multiplied by about 1.1.

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QUESTION 3

GIVEN: Questions on external dosimetry and table of 6 different LiF chips.

SOLUTIONS AND ANSWERS(•):

A. The ICRP Publication 60 equivalent dose **H** is calculated from external tissue absorbed doses: $\mathbf{D_{\beta}}$ of **30 mrad**, $\mathbf{D_{\gamma}}$ of **70 mrad**, $\mathbf{D_{t\text{-}n}}$ of **90 mrad**, and $\mathbf{D_{f\text{-}n}}$ of **25 mrad**:

• $$\boldsymbol{H} = \sum_R w_R\, D_R = (1)(30) + (1)(70) + (5)(90) + (20)(25) = \mathbf{1{,}050}\ \boldsymbol{mrem},$$

where D_R is the absorbed dose for radiation R and w_R is the radiation weighting factor for respectively β, γ, t-n, and f-n radiation. The value of $w_{f\text{-}n}$ is conservatively taken to be 20 for the mixed fast neutron field.

• B. The lifetime dose of 32 rem for a 26 year old worker (age in years = n = 26) exceeds by 6 rem the lifetime recommended limit of **n rem** in NCRP Report 91. Based on recommendations in NCRP Report 91, this worker's allowed annual dose should be limited to 1 rem instead of the 5 rem for workers who do not exceed their lifetime limit.

• C. The allowed dose, per recommendations in ICRP 60, in year 5 of a worker who has received a total dose of 10 rem in the previous 4 years would be zero. The ICRP recommends a dose limit of 5 rem per year with the further restriction that the average annual dose over any 5 year control period does not exceed 2 rem per year or a total of 10 rem, which the worker already received in the first 4 years of the 5 year control period.

• D. For a worker using a Pu-Be fast neutron source, the dosimeter should contain two chips: ^{7}LiF (number 5) and ^{6}LiF(number 6), but number 5 also should have a Cd filter above the chip like that for number 6. The ^{6}LiF chip under the Cd cover responds to albedo neutrons plus external gamma radiation and gamma radiation produced by thermal neutrons absorbed in the Cd filter. The ^{7}LiF under the added Cd filter responds to external gamma radiation and gamma radiation produced by thermal neutrons absorbed in the Cd filter, and its response should be subtracted from the response of the ^{6}LiF chip to obtain the net neutron response of the ^{6}LiF chip.

• E. For an x-ray technologist, a three element badge consisting of three ^{7}LiF chips (numbers 2, 3, and 5) would be desirable for interpretation of skin dose (number 3), eye dose (number 5), and deep dose (number 2). For the premise of measuring effective dose equivalent, the single dosimeter with a 1,000 mg cm^{-2} cover (number 2) is adequate.

• F. Answer is 4.

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QUESTION 4

GIVEN: Neutron activation of a thin gold foil and required data:

T_1	≡	irradiation time = 6 h;
$\mathbf{T_2}$	≡	time after irradiation to beginning of count = **11 h**;
m	≡	mass of gold foil = 0.025 g;
$\boldsymbol{\sigma_{act}}$	≡	activation cross section = **98.8x10^{-24} cm^2 at^{-1}**;
$\mathbf{E_\gamma}$	≡	gamma energy = **0.412 MeV**;
$\mathbf{Y_\gamma}$	≡	gamma yield = **0.955 γ d^{-1}**;
$\boldsymbol{\lambda}$	≡	decay constant = (ln 2)/(2.695)(24 h) = **0.0107 h^{-1}**;
C	≡	net counts in peak = **827,410 c**;
$\mathbf{T_3}$	≡	counting interval = 1 minute = **60 s**; and
ϵ_γ	≡	gamma peak detection efficiency = **0.273 c γ^{-1}**.

SOLUTIONS AND ANSWERS(•):

A. Activity $\mathbf{A(T_1)}$ in Bq when foil is removed from reactor is calculated:

$$\langle C\rangle \;=\; A(T_1)\; e^{-\lambda T_2}\; Y_\gamma\; \epsilon_\gamma\; T_3;\; so$$

• $$\boldsymbol{A(T_1)} \;=\; \frac{C\; e^{\lambda T_2}}{Y_\gamma\; \epsilon_\gamma\; T_3} \;=\; \mathbf{59{,}500}\;\; \boldsymbol{Bq}.$$

B. Gamma exposure rate $\dot{\mathbf{X}}$ in mR h^{-1} at distance **d** of **0.1 m** from an activity **A** of **3.5 mCi** is approximated:

• $$\dot{X} \;\approx\; 0.5\; \frac{A\; Y_\gamma\; E_\gamma}{d^2} \;=\; \mathbf{68.9}\;\; \boldsymbol{mR\;\; h^{-1}}.$$

C. The saturation activity **A** in Bq in a foil with **N** of **1x10^{24} atoms** exposed to a thermal fluence rate $\boldsymbol{\phi}$ of **1x10^{11} n cm^{-2} s^{-1}** is calculated:

• $$\boldsymbol{A} \;=\; \sigma_{act}\; \phi\; N \;=\; \mathbf{9.88}\boldsymbol{x10^{12}}\;\; \boldsymbol{Bq;\;\; so\;\; answer\;\; is\;\; 2}.$$

SOLUTIONS AND ANSWERS TO 1998 ABHP EXAM

QUESTION 5

GIVEN: Neutron detection.

SOLUTIONS AND ANSWERS(•):

• A. The nuclear reactions in the following detectors are specified:

$^{10}BF_3$ counter: **$^{10}B(n,\alpha)^7Li$**; ^6Li counter: **$^6Li(n,\alpha)^3H$**; and ^3He counter: **$^3He(n,p)^3H$**.

• B. To determine the flux (i.e., fluence rate) and average energy using the Bonner sphere method with a LiI(Eu) scintillator, several polyethylene spheres (commonly about 5) ranging in size from about 2 inches to 18 inches in diameter are exposed (usually individually) in the field of interest. The events in the scintillator (from the reaction $^6Li(n,\alpha)^3H$) centered in the sphere are detected by a photomultiplier tube and fed to a preamp-amplifier-scaler and recorded as counts. The calculated counting rates from the different spheres are used together with a predetermined counting efficiency versus neutron energy matrix in a computer interactive program to generate the shape of the energy distribution (An initial guess as to the spectral shape is input by the operator.). From the unfolding routine, the energy distribution can be obtained in the form of fluence rate per unit energy interval, $\phi(E)$. The total fluence rate can be obtained by integrating the distribution over the entire energy range. The average energy, $\bar{E}$, can be obtained by using the fluence rate per unit energy distribution, $\phi(E)$, in a numerical or graphical interpretation when it is noted for classical neutrons that $\phi(E) \equiv n(E)\, v$ and $v = (2\, E/m)^{1/2}$ where n(E) defines the neutron density, v the neutron's velocity, and m the neutron's rest mass:

$$\bar{E} \equiv \frac{\int_0^{E_{max}} n(E)\; E\; dE}{\int_0^{E_{max}} n(E)\; dE} = \frac{\int_0^{E_{max}} \frac{\phi(E)}{v}\; E\; dE}{\int_0^{E_{max}} \frac{\phi(E)}{v}\; dE} = \frac{\int_0^{E_{max}} \phi(E)\; E^{\frac{1}{2}}\; dE}{\int_0^{E_{max}} \phi(E)\; E^{-\frac{1}{2}}\; dE},$$

or by using finite energy bins of width ΔE:

$$\bar{E} = \frac{\sum_0^{E_{max}} \phi(E)\; E^{\frac{1}{2}}\; \Delta E}{\sum_0^{E_{max}} \phi(E)\; E^{-\frac{1}{2}}\; \Delta E}.$$

- C. Foil activation can be used to evaluate the neutron fluence rates and shape of the energy distribution. Foils having particular threshold neutron-induced reactions above a specific neutron energy are selected, and the radioactivities of products are determined through measurements of their radiation emission rates. Using a computer, interactive, unfolding routine, it is possible to determine the shape and fluence rate of the neutron energy distribution from the induced activities in the foils. One advantage of this method is that it is passive and lends itself as a personal criticality dosimetry, which can be measured immediately after a criticality accident to identify persons who may have received significant doses. Because the foils are evaluated after irradiation, no electronics or readout are necessary during irradiation.

- D. The isotope ^{235}U provides for the detection of thermal neutrons by the fission process. Thermal neutrons are not capable of inducing fission in ^{238}U, which has a neutron energy fission threshold of about 1 MeV.

QUESTION 6

GIVEN: Six, two digit, integer counting rates $\mathbf{R_i}$ of a calibration source; experimental mean rate $\bar{R}$ of **63.5 cpm**; experimental estimate of the population standard deviation, $\hat{\sigma}_{\mathbf{exp}}$, of **5.6 cpm**, which can be calculated from the given R_i values as **5.61 cpm** to three significant digits; and table of χ^2 values.

SOLUTIONS AND ANSWERS(•):

• A. A Type I error is falsely concluding activity is present in a sample above the established *physically significant activity* (PSA), which is calculated from the quotient, L_c/E, of the established net counting rate *critical level*, L_c, and counting efficiency, E, and which will occur with a probability α, typically chosen at 5%, when in fact the activity is actually zero. A Type II error is falsely concluding that the activity is less than the established PSA when in fact activity is present above the PSA. When activity is present at the established *minimum detectable activity* (MDA), which is calculated from the quotient, L_d/E, where L_d is the established lower limit of detection for the net counting rate, then there will be a probability β, typically chosen at 5%, of falsely concluding activity is less than the PSA, which is the decision tool used to decide the absence or presence of activity in a sample.

• B. Blank samples are representations of the media and container in which actual radioactivity in samples are counted, but they contain no net activity from the medium sampled, e.g., the air in the working environment of workers sampled with a glass fiber filter. Because blank samples, e.g., new glass fiber filters, may contain varying amounts of radioactivity, the blanks or filters themselves will cause a variation in the observed gross background rate that should be subtracted from the gross rate observed for samples in calculating the net counting rate for a sample. For the filter example, the same filter and planchet used to collect and then count an air filter sample should be counted prior to sampling the air to determine the gross background counting rate, but this procedure often is not followed.

Comment: Unlike the implication in the underlined portion of the question: "What are "blank" samples and why are they used to determine instrument background?", the contribution to the *gross* background rate made by radioactivity present in a blank sample should **not** be considered as part of the instrument background. From a practical standpoint, the same medium used to collect a sample may not be available to use as part of the blank, e.g., chemical reagents, baseline bioassay samples from workers, etc. In such cases, it is more appropriate to first determine the instrument background for establishing the instrument critical level or net counting rate for deciding if **any** activity is present in a sample above that implied by the instrument background. The actual distribution of radioactivity in different blanks, e.g., the distribution of ^{232}Th in baseline fecal samples from a worker population, is then used to establish the PSA above which the decision is made that activity is present above the activity of blanks or baseline samples. The activity in blanks or baseline samples often will be described best by a log-normal distribution whose parameter estimates should be used to establish the PSA for a specified probability α of being exceeded when no activity above the blanks or baseline samples is actually present.

C. The activity **A** and standard error estimate $\hat{\sigma}_A$ are calculated from counts of the sample and an assumed appropriate blank given:

R_{s+b} ≡ gross counting rate = C_{s+b}/T_{s+b} = 500 c/10 min = **50 cpm**;
T_{s+b} ≡ counting interval for sample = **10 min**;
R_b ≡ counting rate for blank = C_b/T_b = 460 c/60 min = **7.67 cpm**;
T_b ≡ counting interval for blank = **60 min**; and
E ≡ gross counting efficiency = **0.15 c d^{-1}**.

- $$\boldsymbol{A} = \frac{R_{s+b} - R_b}{E} = \mathbf{282\ \mathit{dpm}},\ \mathit{and}$$

- $$\hat{\sigma}_A = \left(\frac{1}{E}\right)\left(\frac{R_{s+b}}{T_{s+b}} + \frac{R_b}{T_b}\right)^{1/2} = \mathbf{15.1\ \mathit{dpm}}.$$

- D. The purpose of the chi-square test of a counter is to determine from repetitive counts C_i of a **constant source** over the same counting interval T whether or not the observed experimental estimate of the population variance, $\hat{\sigma}^2_{exp}$ (commonly designated as s^2), is significantly less or more than the theoretical estimate of the population variance, $\hat{\sigma}^2_p$, which is estimated from the sample mean count $\bar{C}$ on the basis of the assumption that the counts have a Poisson distribution approximated by a normal distribution when $\bar{C}$ exceeds about 30 counts. Similar comparisons apply to the counting rates calculated from the counts observed over a constant counting interval. A χ^2 value of 4.01 for 5 degrees of freedom ν is very close to the most likely value (ν-1) of 4 when it is assumed that the counter shows no less and no more variance than expected. The χ^2 value of 4.01 is considerably less than the critical high value of 11.07 shown in the table for deciding that the counting results show too much variance at a significance level of 5%, which is the probability of falsely concluding that the counter is not working properly when it actually is working.

E. An estimate $\hat{\chi}^2$ for the chi-square value cannot be calculated from the given data, because the counting interval T used to determine each of the six counting rate measurements was not given. Two alternative solutions based upon different assumptions are given as follows.

1. If it is assumed that the counting rates were calculated from counts observed over a constant counting interval T of 1 minute (Note: integer values for the reported rates imply a counting interval of 1 minute.), then $\boldsymbol{\hat{\chi}^2}$ can be calculated from the given experimental standard

deviation $\hat{\sigma}_{\mathbf{exp}}$ of **5.6 cpm** and the propagated, theoretical Poisson estimate $\hat{\sigma}_{\boldsymbol{p}}$ of **7.97 cpm** for the population standard deviation, which is calculated from $(\bar{R}/T)^{1/2}$ for the given mean rate $\bar{\boldsymbol{R}}$ of **63.5 cpm** and the assumed counting interval **T** of **1 minute**. The estimated reduced chi-square statistic, $\hat{\chi}_{\nu}^{\mathbf{2}}$, and estimated chi-square statistic, $\hat{\chi}^{\mathbf{2}}$, are calculated then for the value $\boldsymbol{\nu}$ of (n-1) or **5**:

$$\hat{\chi}_{\nu}^{2} = \left(\frac{\hat{\sigma}_{\text{exp}}}{\hat{\sigma}_{p}}\right)^{2} = \left(\frac{5.6}{7.97}\right)^{2} = \mathbf{0.494};\ \textit{so}$$

- $$\hat{\chi}^{2} = \nu\ \hat{\chi}_{\nu}^{2} = (5)(0.494) = \mathbf{2.47}.$$

2. If it is assumed instead that the given standard deviation of **5.6 cpm** is $\hat{\sigma}_{\boldsymbol{p}}$, the Poisson theoretical estimate of the population standard deviation, then the six counting rates can be used to calculate the experimental estimate of the population standard deviation $\hat{\sigma}_{\mathbf{exp}}$ as **5.61 cpm**. The estimated reduced chi-square statistic, $\hat{\chi}_{\nu}^{\mathbf{2}}$, and estimated chi-square statistic, $\hat{\chi}^{\mathbf{2}}$, are calculated then for the value $\boldsymbol{\nu}$ of (n-1) or **5**:

$$\hat{\chi}_{\nu}^{2} = \left(\frac{\hat{\sigma}_{\text{exp}}}{\hat{\sigma}_{p}}\right)^{2} \equiv \left(\frac{5.6}{5.61}\right)^{2} = \mathbf{0.996};\ \textit{so}$$

- $$\hat{\chi}^{2} = \nu\ \hat{\chi}_{\nu}^{2} = (5)(0.996) = \mathbf{4.98}.$$

Comment: This part to the question had no answer based on the given information. The solution in 2 represents a blunder in the application of the chi-square statistic where, in reality, the experimental standard deviation is used incorrectly for $\hat{\sigma}_{\boldsymbol{p}}$. Thus, $\hat{\chi}_{\nu}^{\mathbf{2}}$ is actually being calculated from $\boldsymbol{(\hat{\sigma}_{\mathbf{exp}}/\hat{\sigma}_{\mathbf{exp}})^{2}}$, which yields unity or the expected value, a meaningless result when calculated this way. It is not really appropriate to use the chi-square statistic for only 5 degrees of freedom ν. The expected value μ for the chi-square statistic is ν and its variance σ^2 is 2ν. The *coefficient of variation* (CV), i.e., 100% σ/μ) for χ^2, thus is calculated from100%$(2/\nu)^{½}$ or as 63.2% for 5 degrees of freedom ν. Such a poor precision in the chi-square statistic would not provide a very reliable chi-square test of a counter. For further information about the proper application of the chi-square test, see paper, "Basic Applications of the chi-Square Statistic Using Counting Data," by Mark Tries *et al* in Health Physics 77: pages 441-453, October, 1999. Further discussion relating to the proper application of the chi-square statistic can be found in the comment at the end of the solution to Question 8 on the 1997 ABHP exam.

QUESTION 7

GIVEN: Accelerator facility with average fluence rates, ϕ_R, in units $\mathbf{cm^{-2}\ s^{-1}}$, in control room of specified radiations **R**,: ϕ_γ of **3,000** from 3 MeV photons, $\phi_{t\text{-}n}$ of **500** from thermal neutrons, and , $\phi_{f\text{-}n}$ of **800** from fast neutrons; tables for given radiations **R** of effective dose per unit fluence conversion factors, $\langle H_E/\Phi\rangle_R$ in units of $\mathbf{Sv\ cm^2}$: $\mathbf{8.2x10^{-12}}$ for 3 MeV photons, $\mathbf{2.3x10^{-12}}$ for thermal neutrons, and $\mathbf{1.2x10^{-10}}$ for fast neutrons; table of dose attenuation coefficients, Σ_R in $\mathbf{cm^{-1}}$, for given radiations **R**, assumed for concrete: **0.08** for 3 MeV photons, **0.25** for thermal neutrons, and **0.08** for fast neutrons; and

- $\mathbf{N}$ ≡ number of persons exposed in control room = **10 persons**;
- $\mathbf{T_Y}$ ≡ annual exposure time in control room = 2000 h = $\mathbf{7.2x10^6\ s}$;
- $\mathbf{S}$ ≡ control room wall surface area = $\mathbf{100\ m^2}$;
- $\mathbf{x_w}$ ≡ thickness of control room wall = **2 m**;
- $\mathbf{C}$ ≡ volume cost of concrete = $\mathbf{\$500\ m^{-3}}$; and
- $\mathbf{C_{ALARA}}$ ≡ ALARA cost justification = $2,000/person-rem = **$200,000/person-Sv**.

SOLUTIONS AND ANSWERS(•):

A. Annual collective effective dose, $\mathbf{H_{C\text{-}E}}$, of persons in control room is calculated:

• $$\boldsymbol{H_{C-E}} = N\ T_Y \sum_R \phi_R \left\langle \frac{H_E}{\Phi} \right\rangle_R = \mathbf{8.77}\ \boldsymbol{person-Sv}.$$

B. The concrete wall shield thickness **x** (assumed in addition to that in Part A) that is needed to reduce the $\mathbf{\dot{H}_E}$ in the control room from $\mathbf{100\ mrem\ h^{-1}}$ to $\mathbf{1\ mrem\ h^{-1}}$ is calculated for a buildup factor B of 1:

Thermal neutrons are not likely to make a significant contribution to the shielded dose because of the large value for $\mathbf{\Sigma_{t\text{-}n}}$ of $\mathbf{0.25\ cm^{-1}}$ for thermal neutrons compared to the value of $\mathbf{0.08\ cm^{-1}}$ for both the 3 MeV photons and fast neutrons. The calculation for **x** can be made for the gamma photons and fast neutrons only as verified by the fraction $\mathbf{F_{t\text{-}n}}$ of the 100 mrem h^{-1} and effective dose that is due to thermal neutrons when it is assumed the radiations have the same initial fractional contributions as in Part A:

$$\boldsymbol{F_{t-n}} = \frac{N\ T_Y\ \phi_{t-n} \left\langle \frac{H_E}{\Phi} \right\rangle_{t-n}}{N\ T_Y \sum_R \phi_R \left\langle \frac{H_E}{\Phi} \right\rangle_R} = \frac{0.0828\ person\ Sv}{8.77\ person{-}Sv} = \mathbf{0.00944}.$$

Thus, the initial effective dose equivalent rate, $(\dot{\mathbf{H}}_{\mathbf{E}})_{\mathbf{t-n}}$, from thermal neutrons is calculated as (100 mrem h^{-1})(0.00944) or **0.944 mrem $\mathbf{h^{-1}}$**. The required dose transmission **T** of the additional shield **x** for photons and fast neutrons thus is calculated as (1)/(100 - 0.944) or **$\mathbf{1.01x10^{-2}}$**. The additional shield thickness **x** is then calculated:

- $$\boldsymbol{x} = \frac{1}{\Sigma_R} \ln\left(\frac{1}{T}\right) = \frac{1}{0.08\ cm^{-1}} \ln\left(\frac{1}{0.0101}\right) = \mathbf{57.4}\ \boldsymbol{cm}.$$

- C. It is effective to reduce the annual collective effective dose from **10 person-Sv** to the value of **0.01 person-Sv** based on ALARA:

  The required additional shield thickness **x** is calculated as in Part B except it is assumed that the initial contribution of thermal neutrons is negligible; so the required dose transmission **T** of the additional shield **x** for photons and fast neutrons thus is calculated as (0.01)/(10) or **0.001**, and **x** is calculated:

$$\boldsymbol{x} = \frac{1}{\Sigma_R} \ln\left(\frac{1}{T}\right) = \frac{1}{0.08\ cm^{-1}} \ln\left(\frac{1}{0.001}\right) = 86.3\ cm = \mathbf{0.863}\ \boldsymbol{m}.$$

  The additional concrete cost is calculated from the product **x S C** as **$43,200**. The **ALARA** justified cost is calculated from the product of the collective effective dose saving of 9.99 person Sv and $\mathbf{C_{ALARA}}$ of **$200,000/person-Sv** or as **$2.00 million**. Thus, the ALARA justified cost greatly exceeds the actual cost of the additional concrete.

- D. Six events/conditions that could lead to unusual exposures include:

1. An increase in the beam current would lead to proportional increases in the dose rates in all areas.
2. An increase in the particle beam energy would produce more and higher energy neutrons and possibly more gamma radiation, thereby causing higher dose rates in all areas.
3. A redirection of more of the beam from primary to secondary areas would increase dose rates in secondary areas.
4. Reduction of the beam directed to secondary areas would result in higher beam current and

dose rates in primary areas.

5. Alteration or removal of local shielding in primary or secondary areas may result in higher doses in either area.
6. Violation or the interruption of interlocks could result in high exposures if personnel gained access to areas protected by interlocks.

Comment: Various parts to this question use "effective dose" (ICRP 60 dose term) rather than "effective dose equivalent" (ICRP 26 dose term). What was really meant to be used? Current regulations in the United States are based on ICRP 26 and ICRP 30; so "effective dose equivalent" should be used unless questions specifically address ICRP 60 and other ICRP publications that use effective dose terminology. Regardless, dose terms in a question should be made clear; a candidate should not be forced to guess what is meant. See comment to the Part C solution in Question 8 below.

SOLUTIONS AND ANSWERS TO 1998 ABHP EXAM

QUESTION 8

GIVEN: Water in a pond uniformly contaminated with ^{106}Ru that causes intakes by a raccoon, mice, and persons drinking water from a pond:

$\mathbf{C_w}$	≡	constant water concentration = **250 pCi L^{-1}**;
$\boldsymbol{\lambda}$	≡	decay constant = (ln2)/(1.02)(365) = **1.86x10^{-3} day^{-1}**;
$\mathbf{\bar{E}_\beta}$	≡	average beta energy = (0.039 MeV)/(3) = **0.0130 MeV**;
ALI	≡	ingestion ALI [(ICRP 30)(Note: also equals stochastic ALI)] = **200 µCi**;
$\mathbf{M_R}$	≡	body mass of raccoon = **8 kg**;
$\mathbf{W_R}$	≡	raccoon water intake rate of (0.08)(8) = **0.64 L day^{-1}**;
$\mathbf{k_{eff}}$	≡	effective loss rate constant for raccoon = **0.069 day^{-1}**;
$\mathbf{I_R}$	≡	raccoon mice ingestion rate = (2)(8)/(20) = **0.8 mice day^{-1}**; and
$\mathbf{q_m}$	≡	mouse body burden = **20 pCi mouse^{-1}**.

SOLUTIONS AND ANSWERS(•):

A. The activity concentration $\mathbf{C_R(t)}$ in **pCi kg^{-1}** after a time **t** of **365 days** is calculated:

The raccoon's total daily activity ingestion rate $\dot{\mathbf{I}}$ is calculated from ($\mathbf{I_R}$ $\mathbf{q_m}$ + $\mathbf{W_R}$ $\mathbf{C_w}$) or as **176 pCi day^{-1}**; so

• $$\boldsymbol{C_R(t)} = \frac{\dot{I}}{M_R\ k_{eff}} (1 - e^{-k_{eff}\ t}) \cong \frac{\dot{I}}{M_R\ k_{eff}} = \mathbf{319}\ \boldsymbol{pCi\ kg}^{-\mathbf{1}}.$$

• B. Because C_R(t) calculated in Part A is essentially the steady state concentration, the steady state activity **A** in the raccoon is calculated from (319 pCi kg^{-1})(8 kg) or **2,550 pCi**.

C. The committed effective dose [equivalent](CEDE), $\mathbf{H_E}$, for an adult drinking **2.2 L day^{-1}** of water from the pond for **365 days** in a year is calculated:

The total ingestion intake **I** is calculated: (2.2)(365)L(250 pCi L^{-1}) = 201,000 pCi or **0.201 µCi**; so

• $$\boldsymbol{H_E} = I \left(\frac{5{,}000\ mrem}{ALI} \right) = \mathbf{5}\ \boldsymbol{mrem},$$

which compares to a total, annual average, natural background dose of about 300 mrem, including the contribution of radon and its progeny.

Comment: The question asks for "Effective Dose", which is based on ICRP 60 and 61 and a limit of 2,000 mrem, but the given ALI of 200 µCi is based on the ICRP 30 limit of 5,000 mrem *committed effective dose equivalent* (CEDE). The ICRP 61 ALI for ^{106}Ru is 50 µCi. The question should have asked for the CEDE not the "Effective Dose". The ALIs in ICRP 61 are based upon a committed "effective dose" of 2,000 mrem. It also should be noted that 1 mrem of effective dose in ICRP 61 is not equivalent to 1 mrem of effective dose equivalent in ICRP 30 because of the different measures of detriment that were considered in calculating the weighting factors.

D. The biological half-life, $\mathbf{T_b}$, of ^{106}Ru in raccoons is calculated:

- $$T_b = \frac{ln2}{k_{eff} - \lambda} = \mathbf{10.3\ \textit{days}}.$$

SOLUTIONS AND ANSWERS TO 1998 ABHP EXAM

QUESTION 9

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 ^{239}Pu in glove box = **750 g**;
V	≡	room volume = **108 m^3**;
$T_{1/2}$	≡	half-life of ^{239}Pu = 24,100 y = **7.60x10^{11} s**; so
λ	≡	decay constant of ^{239}Pu (ln2)/$T_{1/2}$= **9.12x10^{-13} s^{-1}**;
F	≡	emergency ventilation of room = 7 m^3 min^{-1} = **0.117 m^3 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 ^{239}Pu = **2x10^{-12} Ci m^{-3}**;
u	≡	mean wind speed = **9 m s^{-1}**;
x	≡	site boundary distance = **1,000 m**;

Stability class **A**; 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 ^{239}Pu is calculated:

• $$\boldsymbol{A} = (9.12x10^{-13}\ s^{-1})\left(\frac{750\ g}{239\ g\ mole^{-1}}\right)\left(\frac{6.02x10^{23}\ at\ mole^{-1}}{3.7x10^{10} at\ s^{-1}\ Ci^{-1}}\right) = \mathbf{46.6}\ \boldsymbol{Ci}.$$

• B. Five factors on which accidental criticality depends include: (1) the mass of fissile material, (2) the geometry of the material, (3) the presence of moderator, (4) the presence of reflector, and (5) the presence of neutron absorbing material.

C. The time **t** is calculated for the concentration **C(t)** to equal the **DAC** of **$2x10^{-12}$ Ci m^{-3}** beginning at an initial concentration **C(0)** of **$4x10^{-4}$ Ci m^{-3}** and for the total removal rate constant **k** of **$1.08x10^{-3}$ s^{-1}**:

- $$\boldsymbol{t} \;=\; \frac{1}{k}\,\ln\!\left(\frac{C(0)}{C(t)}\right) \;=\; 17{,}700\ s \;=\; \mathbf{4.92}\ \boldsymbol{h}.$$

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

Q' = release rate = ($4x10^{-4}$ Ci m^{-3})(0.117 m^3 s^{-1})(0.0005) = **$2.34x10^{-8}$ Ci s^{-1}**;
σ_y = cross wind dispersion coefficient at 1,000 m from graph = **210 m**;
σ_z = vertical dispersion coefficient at 1,000 m from graph = **550 m**; and
y = **z** = **0**; so the given equation reduces and yields answer:

- $$\chi \;=\; \frac{Q'}{\pi\ \sigma_y\ \sigma_z\ u}\ e^{-\frac{1}{2}\frac{h^2}{\sigma_z^2}} \;=\; \mathbf{7.16}\boldsymbol{x10}^{-\mathbf{15}}\ \boldsymbol{Ci}\ \boldsymbol{m}^{-\mathbf{3}}.$$

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QUESTION 10

GIVEN: Release of about 2 kg of "nominally pure", class Y plutonium in a plume that passes over town; specific radionuclide data in table; and other bolded data:

S_α ≡ total α specific activity in table = 0.09 Ci g^{-1};
DCF_1 ≡ external dose factor = 2.8 $x10^{-3}$ rem h^{-1} per g m^{-2}; so
F_{ext} ≡ $(DCF_1)/(S_\alpha)$ = 0.0311 rem h^{-1} per α Ci m^{-2};
$\mathbf{F_{int}}$ ≡ inhalation DCF for internal α CEDE = 330 rem μCi^{-1} = **$3.3x10^8$ rem Ci^{-1}**;
$\mathbf{F_B}$ ≡ breathing rate of 20 L min^{-1} = **1.2 m^3 h^{-1}**; and
$\mathbf{F_R}$ ≡ resuspension factor, # Ci m^{-3} per Ci m^{-2} = **10^{-5} m^{-1}**.

SOLUTIONS AND ANSWERS(•):

• A. Two actions which could reduce the dose significantly to the downwind population during the first week following the accident include:

1. Instruct population to remain in closed homes as much as possible.
2. Instruct population to relocate by moving to homes of relatives or friends that are outside the affected area.

B. Given that the relocation PAG of the EPA > 2 rem TEDE in the first year; that residents outdoors during passage of the plume already received an intake resulting in a CEDE between 1.5 rem to 2 rem; and that residents are **likely** to receive an additional projected CEDE of 1.3 rem in the first year after the accident:

• 1. The PAG is a **projected** dose from a nuclear accident or contamination event above which certain protective actions are recommended to prevent or lessen that dose. Committed doses already received from intakes during the passage of a plume are not included as part of any projected dose from a later phase of a contamination event.
• 2. If the PAG is followed, then even residents who were outdoors during the passage of the plume and as a result already received 1.5 rem to 2 rem would not be advised to relocate themselves outside the area because their additional **projected** TEDE is only 1.3 rem, which is less than the PAG of 2 rem for the first year.

Comment 1: Even if the additional projected TEDE for the first year had exceeded the PAG of 2 rem, a candidate might have difficulty making a recommendation to relocate members of the public based upon a projected 2 rem TEDE when they realize that the corrective action guide of the *Environmental Protection Agency* (EPA) for simply taking actions to fix a home having an average radon concentration greater than 4 pCi L^{-1} corresponds to an effective dose equivalent of 1 rem per year for continuous exposure. Although most homes have average concentrations less than 4 pCi L^{-1}, many homes exceed the 4 pCi L^{-1} guide, but these homeowners are not advised by the EPA to relocate themselves when, for example, their corrective actions have been able to reduce the radon to only 20 pCi L^{-1}, which corresponds to an effective dose equivalent of 5 rem per year for continuous exposure based on an ICRP Publication 32 dose factor. It is to be noted that the effective dose equivalent from alpha particles emitted by radon progeny and the effective dose equivalent from alpha particles emitted by plutonium in the incident described in this part of the question are deemed to have the same risk of cancer mortality.

Comment 2: Further information regarding the PAGs can be found in the document EPA 400-R-92-001 of the *Environmental Protection Agency* (EPA), "Manual of Protective Action Guides and Protective Actions for Nuclear Incidents," May, 1992. The PAGs are designated and applied separately to three phases: (1) PAGs for the *Early Phase* deemed to apply to a potential or actual atmospheric release and to last about 4 days since the onset of a nuclear accident and where evacuation and/or sheltering and the use of the blocking agent KI for projected or measured radioiodine releases may be appropriate actions for members of the public: *total effective dose equivalent* (TEDE) of 1-5 rem (where actions should be taken at a projected dose of 1 rem and even less if such actions have low impact and risks to members of the public) and thyroid *committed dose equivalent* (CDE) of 25 rem; (2) PAGs for the *Intermediate Phase* deemed to last 1 year for deposited radioactivity and where relocation and decontamination may be appropriate actions for members of the public: (a) relocation for TEDE $\geq$ 2 rem and for beta dose to skin 50 times higher, and (b) application of simple dose reduction techniques for TEDE < 2 rem; (3) *Radiation Protection Criteria for the Late Phase* deemed to apply between 1 and 50 years after a nuclear incident are held in reserve. In addition to these three phases and PAGs, this EPA document includes "Protective Action Guides for the Intermediate Phase (Food and Water)" provided by the *Food and Drug Administration* (FDA): (a) *Preventive PAG* where actions having minimal impact are used to reduce radioactive contamination of human food or animal feeds, e.g., storage of food for decay and placing animals on stored feed rather than in contaminated pastures: (i) 1.5 rem projected CDE to the thyroid, or (ii) 0.5 rem projected CEDE to the whole body or 0.5 rem CDE to the bone marrow or any other organ, and (b) *Emergency PAG* where actions are taken to isolate contaminated food to prevent its introduction into commerce and where responsible officials should determine whether condemnation or another disposition is appropriate: 10 times the Preventive PAG: (i) 15 rem projected CDE to the thyroid, or (ii) 5 rem projected CEDE to the whole body or 5 rem CDE to the bone marrow or any other organ. Derived response levels corresponding to these PAGs are calculated from the quotient of the intake corresponding to the PAG and the total consumption for either an adult and the infant as a critical segment of the population.

C. With respect to the calibration of a hand-held, thin-crystal, sodium-iodide based, single channel analyzer for 17 keV and 60 keV photons (Ratio of emission rate of 17 keV to 60 keV photons is approximate 2.5.) to assess surface contamination from the plutonium mix in the given table:

SOLUTIONS AND ANSWERS TO 1998 ABHP EXAM

- 1. Two advantages of each energy calibration when measurements are made for both photons include: (a) The 17 keV calibration provides for estimates of the total activity of plutonium isotopes and ^{241}Am. Because of the much greater attenuation of the 17 keV photons compared to the 60 keV photons, combined measurements of both photon energies will provide for corrections for attenuation in the source material and air when account is taken for the stated emission ratio of 2.5. (b) The 60 keV calibration provides for estimates of the total activity of ^{241}Am. When corrections for attenuation of both energy groups are made, the net response from plutonium isotopes can be obtained by subtracting the predicted 17 keV response for ^{241}Am from the gross 17 keV response, thereby providing an estimate of the total activity of plutonium isotopes.

  Comment: Although the answers shown in 1 are "theoretically" possible based on the premise, field measurements with a sodium iodide detector (which has poor energy resolution and a Compton continuum background in both the 17 keV and 60 keV windows from natural background photon radiation) are not likely to achieve the activity resolutions suggested above.

- 2. The recommended photon energy for measurements:
  a. on paved road surfaces is 17 keV because of its 2.5 fold greater emission rate if equal detection efficiencies for the 17 keV and 60 keV photons are assumed.
  b. on an agricultural field following an extended rain is 60 keV because of the expected much greater attenuation of the 17 keV photons compared to the 60 keV photons.

D. The internal **CEDE** for a person who walks for a time **T** of **1 h** on soil having a surface activity $\mathbf{A_S}$ of 100 µCi m^{-2} or $\mathbf{10^{-4}\ Ci\ m^{-2}}$ is calculated from factors and bolded units:

- $$\boldsymbol{CEDE} = (A_S)(F_R)(F_B)(T)(F_{int}) = \mathbf{0.396}\ \boldsymbol{rem}.$$

- E. Refinements of a TEDE estimate of 4.8 rem for a worker calculated from default assumptions for the facility’s emergency plan are necessary by actions, for example: (1) estimate the worker’s intake from data for that worker’s *personal air sampler* (PAS), which has a *minimum detectable intake* (MDI) of about 0.0005 ALI for ^{239}Pu , (2) if justified by the PAS data, obtain fecal samples over time to estimate the respirable and non-respirable portions of the intake, which have a MDI of about 0.1 ALI for Pu-239, (3) analyze PAS filter samples for activity ratios of radionuclides and the particle size distribution of radioactive aerosols collected on the filter, (4) obtain chest count even though MDI $\geq$ 5,000 ALI for Pu-239 (Detection of 60 keV photons from Am-241 improves intake detection limit for Pu-239 if activity ratio is known), and (5) obtain urine samples even though MDI $\geq$ 5 ALI for Pu239.

QUESTION 11

GIVEN: Data for radiopharmaceuticals used in heart muscle blood perfusion studies: ^{99m}Tc (Tetrolite and designated by subscript 1) and ^{201}Tl (Thallium and designated by subscript 2), where it is stated that the injection of 1 can be 10 times that of 2 without increasing the patient's effective dose equivalent:

- $\mathbf{I_1}$ ≡ total intake of ^{99m}Tc = **30 mCi**;
- $\mathbf{I_2}$ ≡ total intake of ^{201}Tl = **4 mCi**;
- **K** ≡ biological removal rate constant of (ln2)/(10 days) = **0.0693 day^{-1}**;
- $\boldsymbol{\lambda_1}$ ≡ ^{99m}Tc decay constant = (ln2)/(6/24) days = **2.77 day^{-1}**; so
- $\mathbf{k_1}$ ≡ ^{99m}Tc total removal rate constant = λ_1 + K = **2.84 day^{-1}**;
- $\boldsymbol{\lambda_2}$ ≡ ^{201}Tl decay constant = (ln2)/(73/24) days = **0.228 day^{-1}**; so
- $\mathbf{k_2}$ ≡ ^{201}Tl total removal rate constant = λ_2 + K = **0.297 day^{-1}**; and
- **m** ≡ mass of total body of **70,000 g**.

SOLUTIONS AND ANSWERS(•):

• A. The three major factors used to *"define"*, i.e., to determine, the CEDE are: (1) the activity **I** injected; (2) the total effective removal rate constant **k** whose inverse gives the mean residence time $\bar{t}$ in the body and which is used to calculate the ultimate number of transformations or cumulated activity, $\tilde{\mathbf{A}}$, from the quotient **I/k**; and (3) the *specific effective energy*, **SEE(WB←WB)**, which is the energy in MeV absorbed per gram of the whole body (WB) per transformation within the WB since the quality factor and tissue weighting factor in this case are both unity (This is the MIRD S factor divided by 2.13.).

B. The relative contribution, **R** (factor for ^{201}Tl (2) relative to factor for ^{99m}Tc (1), of each factor in determining the nearly10 fold difference in effective dose equivalent per mCi are calculated:

From the values shown in the table, the SEE factors are calculated:

$$SEE_1 = \frac{(0.1405\ MeV)(0.891\ d^{-1})(0.36)}{70{,}000\ g} = 6.44x10^{-7}\ MeV\ g^{-1}\ d^{-1}.$$

$$SEE_2 = \frac{[(0.0689)(0.27) + (0.0708)(0.465) + (0.0803)(0.205)]\ (0.47)\ MeV\ d^{-1}}{70{,}000\ g} = 4.56x10^{-7}\ MeV\ g^{-1}\ d^{-1}.$$

Comment: Actually, the contributions of Auger and internal conversion electrons are not insignificant as stated in the given table, and they should be included.

For factor (1), the intakes $\mathbf{I_1}$ and $\mathbf{I_2}$, the relative contribution in the dose per mCi, $\mathbf{R_{(1)}}$, is by definition unity:

- $$R_{(1)} = \frac{\left(\frac{I_2}{I_2}\right)}{\left(\frac{I_1}{I_1}\right)} = \mathbf{1}.$$

For factor (2), the rate constants $\mathbf{k_1}$ and $\mathbf{k_2}$, the relative contribution in the dose per mCi, $\mathbf{R_{(2)}}$, is calculated:

- $$R_{(2)} = \frac{\left(\frac{1}{k_2}\right)}{\left(\frac{1}{k_1}\right)} = \frac{\left(\frac{1}{0.297}\right)}{\left(\frac{1}{2.84}\right)} = \mathbf{9.56}.$$

For factor (3), the $\mathbf{SEE_1}$ and $\mathbf{SEE_2}$, the relative contribution in the dose per mCi, $\mathbf{R_{(3)}}$, is calculated:

- $$R_{(3)} = \frac{SEE_2}{SEE_1} = \frac{4.56x10^{-7}}{6.44x10^{-7}} = \mathbf{0.708}.$$

Comment: The instruction for calculating the ratio of the factors on the basis of the dose per mCi is confusing, especially for the ratio $\mathbf{R_{(1)}}$ involving the intake factor. In fact, there is only a factor of 6.77 not 10 in the dose per mCi. The ratio **R** of the dose from ^{201}Tl to the dose for ^{99m}Tc for the given intakes is calculated:

$$R = \left(\frac{4\ mCi}{30\ mCi}\right)(9.56)(0.708) = \mathbf{0.903},$$

which agrees with the premise to this question. However, I_1/I_2 only equals 7.5 and not 10. I believe that the ABHP meant to ask for ratios on the basis of relative "doses" and not doses per mCi.

C. The effective half-life, $\mathbf{T_{eff}}$, of ^{99m}Tc in the gall bladder where it has a biological half-life of 7 days (**K** = (ln 2)/(7 days) = $\mathbf{0.0990\ day^{-1}}$) is calculated:

- $$T_{eff} = \frac{\ln 2}{\lambda_1 + K} = \frac{\ln 2}{2.77\ day^{-1} + 0.099\ day^{-1}} = \mathbf{0.242}\ \boldsymbol{days}.$$

D. Given that 20 times more counts are acquired with ^{99m}Tc (1) than with ^{201}Tl (2), i.e., $\mathbf{C_1 = 20\ C_2}$, and that variability $\mathbf{V} \equiv \hat{\sigma}_c\mathbf{/C = C^{1/2}/C = 1/C^{1/2}}$, the relative variability $\mathbf{V_2/V_1}$ is calculated:

- $$\frac{V_2}{V_1} = \frac{\dfrac{1}{C_2^{\ 1/2}}}{\dfrac{1}{C_1^{\ 1/2}}} = \frac{\dfrac{1}{C_2^{\ 1/2}}}{\dfrac{1}{(20\ C_2)^{1/2}}} = 20^{1/2} = \mathbf{4.47}.$$

E. The extremity dose of ^{201}Tl (2) relative to that for ^{99m}Tc (1), $\mathbf{D_2/D_1}$, is calculated by assuming that the tissue energy absorption coefficients for the photons are a constant factor times those given in the table for air and that $\mathbf{T_1 = 2\ min}$ and $\mathbf{T_2 = 1\ min}$:

- $$\frac{D_2}{D_1} = \left(\frac{1}{2}\right)\left(\frac{(68.9)(0.27)(0.0262) + (70.8)(0.465)(0.0262) + (80.3)(0.205)(0.0236)}{(140.5)(0.891)(0.0245)}\right) = \mathbf{0.283}.$$

SOLUTIONS AND ANSWERS TO 1998 ABHP EXAM

QUESTION 12

GIVEN: Gaseous release data for PWRs and BWRs:

Q ≡ acceptable release rate = $\mathbf{8.35x10^{4}}$ **μCi s^{-1}**;
C/R ≡ concentration causing net rate of 1 cpm = $\mathbf{1.69x10^{-6}}$ **μCi cm^{-3} per cpm**;
R_b ≡ background rate = **730 cpm**; and
F ≡ flow rate at release point = 6,300 ft^3 min^{-1} = $\mathbf{2.97x10^{6}}$ **cm^3 s^{-1}**.

Comment: The question improperly describes C/R as the "Detector efficiency".

SOLUTIONS AND ANSWERS(•):

A. The gross counting rate alarm set point, $\mathbf{R_{alarm}}$, for **0.6 Q** is calculated:

• $$\boldsymbol{R_{alarm}} = R_b + \frac{0.6\ Q}{F\ (C/R)} = 730 cpm + 9{,}980\ cpm = \mathbf{10{,}710}\ \boldsymbol{cpm}.$$

• B. Three sources of radioactive gases in the reactor coolant are: (1) neutron activation of dissolved gases, (2) fission gases that leak from the inside of the fuel rods, and (3) fission gases produced by fission of *tramp* uranium on the outside surfaces of the fuel rods.

Comment: The fast neutron reaction, $^{16}O(n,p)^{16}N$, with oxygen in the water molecule also might be included, but the ^{16}N will most likely not be present as a gaseous species. The ^{16}N may be present in a volatile form such as $^{16}NH_3$ when the coolant represents a reducing environment such as under hydrogen injection in a BWR.

• C. Three mechanisms by which tritium is produced in a PWR include: (1) through ternary fission, (2) through activation of deuterium ($^{2}H(n,\gamma)^{3}H$) present in some of the water molecules, which is present in hydrogen at an atom abundance of about 0.015%, and (3) through activation of the chemical shim, boric acid, through the reaction $^{10}B(n,\alpha)^{*7}Li$ and the fission of the excited product $^{*7}Li$ as follows: $^{*7}Li \rightarrow {}^{4}He + {}^{3}H$.

• D. The BWR coolant has lower concentrations of radioactive gases than those in a PWR because gases (including radioactive gases, hydrogen and oxygen) are stripped from the steam when it is condensed back to water in the BWR coolant

• E. Hydrogen is produced by the radiolytic dissociation of the water molecule by the intense radiation field in the reactor core: H_2O + radiation → H_2 + O.

• F Two methods for preventing hydrogen gas from reaching a concentration at the explosive limit are (1) recombination of the H_2 with oxygen to form H_2O and (2) dilution of the hydrogen concentration with air before it reaches the lower explosive limit.

- G With respect to a particular radionuclide in a BWR:

1. The radionuclide ^{16}N produces the highest external dose rate during reactor operation.
2. The highest accessible dose rate likely would be at contact with the steam line leading from the containment building to the turbine building.
3. It is not a problem after shutdown because it quickly decays with a very short half-life of about 7 seconds.
4. The production mechanism is the fast neutron reaction, $^{16}O(n,p)^{16}N$, with oxygen in the water molecule.

SOLUTIONS AND ANSWERS TO 1998 ABHP EXAM

QUESTION 13

GIVEN: A technician concerned about exposure from ^{125}I in a dialysis unit:

d	≡	distance from unit = **0.45 m**;
A(0)	≡	initial activity = **370 Mbq**;
t	≡	time to beginning of exposure = **14 days**;
T	≡	exposure interval = **1.5 h**;
λ	≡	^{125}I decay constant = (ln 2)/(60.1 days) = **0.0115 day^{-1}**;
μ	≡	attenuation coefficient for lead = (28.9 cm^2 g^{-1})(11.35 g cm^{-3}) = **328 cm^{-1}**; and
Γ	≡	^{125}I gamma constant = **7.4x10^{-5} mSv h^{-1} Mbq^{-1} m^2**.

SOLUTIONS AND ANSWERS(•):

• A. Two routes of exposure and their assessment include: (1) external radiation assessed by measurement with a survey meter or dosimeter badge and (2) internal exposure assessed by counting the technician's thyroid with a thin sodium iodide detector.

B. The technician's effective dose [equivalent] $\mathbf{H_E}$ is calculated:

•
$$\boldsymbol{H_E} = \left(\frac{\Gamma\ A(0)\ e^{-\lambda t}}{d^2} \right) T = 0.173\ mSv = \mathbf{17.3\ \textit{mrem}}.$$

Comment: For the approximate 30 keV photons of ^{125}I, the calculated dose equivalent is near the surface of the body. The actual effective dose equivalent to the whole body would be considerably less and is not calculated from the given information.

C. The dose equivalent rate $\dot{\mathbf{H}}_\mathbf{E}$ at a distance **d** of **0.45 m** from an activity **A** of **200 MBq** that is shielded with lead having a thickness **x** of **0.015 cm** is calculated by assuming that all photons have the same linear attenuation coefficient **μ** of **382 cm^{-1}** in lead and that buildup need not be considered because most of the interactions in lead will be by the photoelectric effect, which leads to complete absorption of the low energy photons:

•
$$\dot{\boldsymbol{H}}_{\boldsymbol{E}} = \left(\frac{\Gamma\ A}{d^2} \right) e^{-\mu x} = 5.33x10^{-4}\ mSv\ h^{-1} = \mathbf{0.0533\ \textit{mrem}\ \textit{h}^{-1}}.$$

• D. Two other actions besides shielding to lower the technicians dose include: (3) place dialysis unit at a greater distance from the sink where cell washes are made and (4) have technician use a different sink to perform cell washes.

SOLUTIONS AND ANSWERS TO 1998 ABHP EXAM

QUESTION 14

GIVEN: Professor using diode laser pointer and your analysis per ANSI Z136.1-1993:

- $\mathbf{\Phi}$ ≡ output power = **4.5 mW**;
- **a** ≡ aperture diameter = **0.2 cm**;
- $\boldsymbol{\phi}$ ≡ beam divergence of 0.2 milliradian = **0.0002 radian**;
- **d** ≡ distance from students = **300 cm to 5,000 cm**;
- $\mathbf{T_a}$ ≡ blink aversion time = **0.25 s**;

Equations for *maximum permissible exposure* (MPE) and *nominal ocular hazard distance* (NOHD).

SOLUTIONS AND ANSWERS(•):

A. The emerging irradiance $\mathbf{I_e}$ in $\mathbf{mW\ cm^{-2}}$ is calculated:

• $$I_e = \frac{\Phi}{\pi \left(\frac{a}{2} \right)^2} = \mathbf{143\ mW\ cm^{-2}};\ \textit{so answer is } \mathbf{d}.$$

B. The **MPE** in $\mathbf{mW\ cm^{-2}}$ is calculated from the given equation, modified for desired units and simplified, where t(s) in the denominator on the RHS of the given equation is interpreted as the given blink aversion time $\mathbf{T_a}$ of **0.25 s**:

• $$MPE = 1.8\ t^{-0.25} = \mathbf{2.55\ mW\ cm^{-2}};\ \textit{so, answer is } \mathbf{c}.$$

• C. The NOHD is the intrabeam axial distance from the laser to the exposed individual's eye beyond which the exposure would be less than the MPE.

D. The **NOHD** is calculated for the given **MPE** of $\mathbf{3\ mW\ cm^{-2}}$ from the given equation from the ANSI standard:

• $$NOHD = \left[\frac{1}{\phi} \right] \left\{ \frac{4\ \Phi}{\pi\ (MPE)} - a^2 \right\}^{\frac{1}{2}} = 6{,}840\ cm = \mathbf{68.4\ m}.$$

• In normal use the laser would not be aimed at the audience; so the eye would not be located along the beam axis and the NOHD therefore would not apply.

- E. The laser would fall in class 3 a (visible, CW output > 1 mW but < 5 mW).

- F. Recommended precautions for the professor include:

1. Avoid directing laser beam into the audience.
2. Avoid directing laser beam at specularly reflecting surfaces, e.g., window glass.

- G. The HeNe laser is lower power (0.5 mW to 4.5 mW) than the diode laser. Beam divergence is probably somewhat higher than the 0.2 milliradian specified for the diode laser. The NOHD would likely be less for the Class II HeNe laser than for the diode laser.