

INTERNAL DOSIMETRY—A REVIEW

Charles A. Potter*

Abstract—The field history and current status of internal dosimetry is reviewed in this article. Elements of the field that are reviewed include standards and models, derivation of dose coefficients and intake retention fractions, bioassay measurements, and intake and dose calculations. In addition, guidance is developed and provided as to the necessity of internal dosimetry for a particular facility or operation and methodology for implementing a program. A discussion of the purposes of internal dosimetry is included as well as recommendations for future development and direction.

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INTRODUCTION

IT IS A PRIVILEGE to be chosen to write this review article on internal dosimetry. What follows is a review of the subject including some of the historical standards, the genesis and evolution of biokinetic models, and methods of intake and dose estimation. Most of this information has been published and promulgated through International Commission on Radiation Protection (ICRP) publications, which are cited frequently. The very early information (including some published concurrently or subsequently by the ICRP) was published as National Bureau of Standards (NBS—now NIST) Handbooks as generated by the National Council on Radiation Protection and Measurements (NCRP). NBS handbooks evolved into what are now NCRP publications and are another source for internal dosimetry information. The information presented in this paper is the basis for a course in internal dosimetry taught by the author. It is hoped that readers will find it useful.

* Sandia National Laboratories, PO Box 5800, MS0651, Albuquerque, NM 87185.

For correspondence or reprints contact the author at the above address, or email at capotte@sandia.gov.

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DOSIMETRY VS. INTERNAL DOSIMETRY

Many of the concepts that are connected to the term *dosimetry* are lost when one considers *internal dosimetry*. Dosimetry quantities and units are defined by the International Commission on Radiation Units and Measurements in publications such as ICRU Report 51, *Quantities and Units in Radiation Protection Dosimetry* (ICRU 1993). These reports remind us that we need to be concerned with quantities like energy imparted, linear energy transfer, lineal energy, absorbed dose, etc. Considerations of the distribution of energy absorbed or charged particles over the dose volume are extremely important in the determination of these quantities.

There are three basic assumptions made in internal dosimetry. First, is that all energy emitted from alpha and beta radiations in an organ or tissue is absorbed in that organ or tissue. Second is that an element deposited in an organ or tissue is instantaneously and uniformly distributed throughout the mass of that organ or tissue. These assumptions very much simplify the calculation of dose to an organ or tissue by setting the absorbed dose in that organ or tissue equal to the total energy deposited divided by the mass with no other considerations. A third assumption is that the rate of change of material in an organ or tissue is proportional to the amount of material in that organ or tissue. The use of this assumption will be discussed later.

Purpose of internal dosimetry

The purpose of internal dosimetry has evolved over time and is best represented as having three objectives. These objectives are listed in their order of importance:

1. **To provide timely feedback on workplace control.** This is listed first because it is the only way that workers can be protected through an internal dosimetry program. A program should include exposure monitoring that provides timely feedback with “special bioassay” initiated by the exposure monitoring results or other workplace indicators. Such a program will result in the mitigation of internal exposure workplace hazards thus providing timely control of the workplace.

2. **To initiate medical intervention.** In cases where large intakes have occurred, medical intervention can be initiated to reduce uptake by systemic organs and tissues and subsequent dose. There are two reasons that this purpose is listed second instead of first. First, medical intervention treatments are commonly initiated before dose data are available and, therefore, are based on other information regarding exposures. This is because they typically are more effective if initiated as soon as possible after intake. The second reason is that intakes requiring medical intervention are relatively rare in comparison to the number and frequency of individuals receiving internal exposure.
3. **To show compliance with a standard or regulation.** Most entities processing or using radioactive materials answer to some authorizing body that has some limit on dose received by individuals. ICRP Publication 60 (ICRP 1991a) refers to limiting "effective dose," but indicates that "The limits apply to the sum of the relevant doses from external exposure in the specified period and the 50-year committed dose . . . from intakes in the same period." The results of internal dose calculations are therefore included as a part of demonstration that limits have been met. Commonly in the United States, regulations of the U.S. Nuclear Regulatory Commission (10 CFR Part 20) or the Department of Energy (10 CFR 835) will apply (NRC 2004; U.S. DOE 2004).

It is vital that all three of these purposes are considered during the development of an internal dosimetry program.

STANDARDS

Early in the history of radiation protection, standards were developed quite differently than they are today. Radionuclides were addressed singularly and limits were based on research with the particular radionuclides. This process took a long time and few standards were actually developed. Below are descriptions of the development of three more widely known standards. For more information on the development of standards, the author recommends *Radioactivity and Health, A History* by J. N. Stannard (1988).

Radium

As most health physicists are aware, internal radiation exposure effects were first noted for painters of radium dials. These individuals, mostly female, developed the habit of twisting their paintbrushes on their lower lips and tongues to get a finer point. This resulted in relatively large ingestions of radium by these individuals. In 1924, a dentist by the name of Blum identified

necrosis of the jaw in some of these individuals (Blum 1929).

As radium ingestion became recognized as the source of these effects, the need for determination of the amount of radium in the body was identified. In 1934, Robley Evans devised and used the meter-arc technique to measure the radium deposition in the individual radium-dial painters. This technique arranged the body of the individual on an arc of 1-m radius with a large Geiger-Mueller tube at the center of the circle formed by the arc. He made four measurements with the front of the individual facing the tube, the back of the individual facing the tube, the attenuation of a known activity ^{226}Ra source attenuated by the body of the individual, and finally the activity of the unattenuated source. With these four measurements, he was able to infer the body content of ^{226}Ra of the individual. Later, to increase the comfort of this extreme procedure, he designed and built the first whole-body counter at the Massachusetts Institute of Technology (MIT). Measurements were made with the patient in a chair and a Geiger-Mueller detector 1 m away (Evans 1937).

Seven years later the radium standard, the first internal dosimetry standard, was promulgated. This "tolerance level," issued in NBS Handbook 27, was a 0.1 μCi "body burden" of ^{226}Ra (NBS 1941). This standard was based on measurements taken at MIT on exposed individuals, many of whom were of radium dial painters (Evans 1981). This standard remained in affect for over 30 years and was the standard on which plutonium and, later, all other bone-seeking radionuclides were based.

Uranium

The second internal emitter standard was for uranium. In the early 1940s, it was recognized that large amounts of uranium would be necessary for use in the Manhattan Project. To determine the effects that uranium might have on the body, experiments were performed at the University of Rochester and at the Metallurgical Laboratory, now Argonne National Laboratory. These experiments showed that chemical toxicity, as opposed to radiological effects, were more limiting for uranium. The chemical toxicity was thought to be similar to lead, for which a standard had been promulgated in 1933. Therefore, in 1944, the maximum permissible concentration in air or (MPC)_a of 150 $\mu\text{g m}^{-3}$ for lead also was adopted for uranium (Spoor and Hursh 1978).

Plutonium

The next important standard to come into existence was that for ^{239}Pu . After plutonium was identified as useful to the Manhattan Project, comparative toxicology studies were performed at the Metallurgical Laboratory

and the University of Rochester. These mostly rodent studies were performed by comparing the effects of radium, polonium, and plutonium. In addition, plutonium was injected into 18 terminally ill patients between 1945 and 1947. In 1950, Austin Brues of the Metallurgical Laboratory proposed the first method by calculation for the determination of a radiological limit. The equation used was:

$$(MPBB)_{Pu} = 0.1 \mu\text{Ci Ra} \left[\frac{1}{15} \times \frac{0.75}{0.25} \frac{4.8 + 0.5(5.5 + 6.0 + 7.7)}{4.8 + 0.15(5.5 + 6.0 + 7.7)} \right] = 0.04 \mu\text{Ci.} \quad (1)$$

The ratios included in this equation were as follows:

- 1/15 = toxicity ratio of plutonium vs. radium in rodent;
- 0.75/0.25 = retention of plutonium vs. radon in rodent; and
- 0.5/0.15 = radon retention in man vs. rodent.

The other values are energies of alpha particles emitted from ^{226}Ra and its progeny. This standard was debated between Brues and Wright Langham of the Los Alamos Laboratory (now Los Alamos National Laboratory) who worked with human data and believed a larger value was more appropriate. Shields Warren, then director of Biology and Medicine at the Atomic Energy Commission, decided in favor of the Brues proposal (Langham and Healy 1978).

SYSTEMS AND MODELS

The decade of the 1950's saw changes in the way internal standards were developed. Rather than choosing possibly arbitrary body burden or airborne concentration limits, methods were developed for modeling the metabolism of radionuclides. These models were used to infer the dose in an individual using default parameters for size and weight of individuals and their organs.

ICRP Publication 2

The first comprehensive publication of internal standards was ICRP Publication 2, *ICRP Report of Committee II on Permissible Dose for Internal Radiation* (ICRP 1959). This publication presented MPC values for radionuclides whose radiologic and metabolic properties were known at that time. The tables included values for concentration in air and water (analogous to the inhalation and ingestion pathways, respectively) for both an occupational (40-h) week and a non-occupational (168-h) week. These values were calculated based on

actual dose limits. Those dose limits, "maximum permissible annual doses," included:

- 3 rems per 13 weeks to the gonads or total body (0.1 rems/week) with a cap of $5(n - 18)$ rem to the total body where n is the age in years;
- an "effective RBE dose" to bone over 13 weeks less than the "average RBE dose" to the skeleton due to a body burden of $0.1 \mu\text{Ci}$ of ^{226}Ra (approximately 0.56 rems/week);
- 4 rems per 13 weeks or 15 rems per year to any single organ except gonads, bone, skin, or thyroid (0.3 rems/week); and
- 8 rems per 13 weeks or 30 rems per year to skin or thyroid (0.6 rems/week).

The concept of a "critical organ" was introduced. The critical organ was the organ or tissue thought to be limiting as to the above dose limits. Typically, several organs and tissues were considered for each radioelement, as well as "soluble" and "insoluble" chemical forms. The critical organ was mathematically treated as a single compartment model where material was introduced on a constant basis (representative of chronic intake) and was removed exponentially. The removal was both radiological and physiological in nature with the physiological aspect being described by a "biological half-life."

The respiratory and gastrointestinal (GI) tracts were also considered. The respiratory tract model took the solubility of the radioelement into account by assuming a fraction would be absorbed into the bloodstream either immediately (soluble) or with a 120-d biological half-life (insoluble). The insoluble biological half-life was used in considering the lung as a critical organ. The GI tract model did not assume exponential removal to the next segment of the GI tract. Instead, a mean residence time was used to model slug flow through the system. GI tract solubility was represented by a parameter, f_1 , which described the fraction of material translocated from the small intestine to the bloodstream. This parameter was only used for ingestion intakes—absorption of material cleared from the lung to the GI tract was not considered.

A "Standard Man" was derived with associated parameters. The information included height, weight, organ radii and masses, breathing rate, and other relevant parameters. The MPCs calculated in the publication were actually those concentrations that would result in a maximum permissible annual dose during the 50th year after 50 y of continuous exposure to standard man.

Other values provided in the publication included "maximum permissible body burdens," "maximum permissible organ burdens," and "effective energies" for each radionuclide. The first two were unfortunately

named because they represented maximum values in the body or organ in the 50th year and only after 50 y of continuous exposure at the MPC. These values were not representative of a “maximum permissible” value at any other time (unless the radionuclide was short lived). The effective energy term represented the product of the energy absorbed in the organ per disintegration, a term representing disintegrations of radioactive progeny, the “relative biological effectiveness,” and a “relative damage factor” for bone-seeking radionuclides.

ICRP Publications 23/26/30

The recommendations, models, and physiological parameters given in ICRP Publication 2 were changed in the 1970's in a series of publications. In 1975, ICRP Publication 23, *Reference Man*, was issued to replace Standard Man (ICRP 1975). ICRP Publication 26 was issued in 1977 and included new dose limits and secondary limits for use in radiation protection (ICRP 1977). ICRP Publication 30 (Part 1) was issued in 1979, and gave detailed information on the calculation of secondary limits for internal dosimetry, a new respiratory tract model, an updated gastrointestinal tract model, a new bone dosimetry model, and metabolic data for many elements (ICRP 1979). The methodology for calculation of secondary limits and doses as well as many of the models presented in this system are still current within the ICRP radiation protection framework.

ICRP Publication 26 included two dose limits for internal exposure. The first limit of 0.5 sievert (Sv) committed dose equivalent was to control non-stochastic effects to any organ or tissue. Committed dose refers to dose accumulated over 50 y after exposure. The second dose limit of 0.05 Sv committed effective dose equivalent was to control stochastic effects to the body. The committed effective dose equivalent quantity was not explicitly named by the ICRP in the original publication, but was written as:

$$\sum_T w_T H_{50,T}. \quad (2)$$

As can be seen by eqn (2), effective dose refers to obtaining the committed dose equivalent to each organ or tissue T ($H_{50,T}$), multiplying those doses by an appropriate weighting factor (w_T), and summing those products. Later, in ICRP Publication 42 (ICRP 1984), the ICRP named the quantity the committed effective dose equivalent. Committed dose equivalent was calculated by the following equation:

$$H_{50,T} = 1.6 \times 10^{-10} \sum_S U_S \sum_i SEE(T \leftarrow S)_i, \quad (3)$$

where U_S = “ultimate” number of disintegrations in source organ S over 50 y after intake, and $SEE(T \leftarrow S)_i$ = “specific effective energy” to target organ T from source organ S per disintegration for radiation i .

The “ SEE factor” could be calculated from its constituents:

$$SEE(T \leftarrow S) = \sum_i \frac{Y_i E_i Q_i AF(T \leftarrow S)_i}{M_T}, \quad (4)$$

where

Y_i = yield of radiation i ;

E_i = characteristic or average energy of radiation i ;

Q_i = quality factor for radiation i ;

$AF(T \leftarrow S)_i$ = absorbed fraction of energy for radiation i in target organ T emitted from source organ S ; and

M_T = mass of target organ T .

Specific effective energy as expressed by eqn (4) replaced the effective energy concept provided in ICRP Publication 2.

Two secondary limits were defined and provided—the “annual limit on intake” or ALI and the “derived air concentration” or DAC. The annual limit on intake was the intake during a year of practice that would result in a committed effective dose equivalent of 0.05 Sv or a committed effective dose to a single organ or tissue of 0.5 Sv to Reference Man. ALIs were calculated based on single acute intakes and were provided for both ingestion and inhalation intakes. DACs were calculated by dividing the ALIs for inhalation intake by the amount of air breathed by Reference Man in one year or 2,400 m³.

No longer was the body represented by a single compartment. The respiratory tract model provided in ICRP Publication 30 contained 10 compartments (including pulmonary lymph nodes), which included translocation from one compartment to another within the respiratory tract, translocation to the gastrointestinal tract, and absorption into the bloodstream referred in the report as the “transfer compartment.” Pathways including absorption from the GI tract of material previously cleared from the lung and swallowed were considered. The standard metabolic model included the transfer compartment out of which material was translocated to named organs or tissues. Material translocated out of these organs and tissues was considered to proceed directly to excretion. Translocation of material out of all compartments including those of the GI tract was described by a simple exponential term. Except in the case of iodine, no recycling of elements was considered.

ICRP Publications 60/61

In 1990, the ICRP issued two publications that included some fundamental changes to the previous philosophy. Changes were made to many of the quantities and units, as is shown in Table 1. Up to this point, committed dose limits were recommended for intakes over 1-y periods. ICRP Publication 60 recommended that committed effective dose be limited over any 5-y period to 0.1 Sv and also be limited to 0.05 Sv per year (ICRP 1991a). A committed equivalent dose limit was no longer considered, because deterministic effects were assumed to be adequately limited by meeting the stochastic dose limit. The values provided for radiation and tissue weighting factors were changed from ICRP Publication 30 including the treatment of the “remainder,” or those organs and tissues not specifically assigned a tissue weighting factor. ICRP Publication 61 contained ALIs and DACs using these recommendations and a reference committed effective dose of 0.02 Sv, representing the 0.1 Sv per 5-y period limit (ICRP 1991b).

ICRP Publications 60/66/56/67/69/70/72/68

The current set of recommendations and models can be found in ICRP Publications 60, 66, 56, 67, 69, 70, 72, and 68. ICRP Publication 60 still contains the most recent recommendations for dose limitation, as well as the current set of quantities and units used in internal dosimetry (ICRP 1991a). ICRP Publications 66, 56, 67, 69, 70, and 72 contain models that replace many of those provided in ICRP Publication 30, although some of the ICRP Publication 30 models are still current (ICRP 1994, 1989, 1993, 1995b, 1995c, 1996). ICRP Publication 68 contains dose coefficients using the current models (ICRP 1995a).

ICRP Publication 66 presents an updated respiratory tract model for use in radiation protection. This model separates individual models for deposition, clearance, absorption, and dosimetry. Deposition is based on both thermodynamic and aerodynamic properties, and the default particle size for inhalation intake has been increased from 1 to 5 μm . Clearance and absorption

pathways compete in each compartment, as opposed to being represented by separate compartments. Default absorption classes (F, M, and S roughly corresponding to D, W, and Y from the earlier ICRP recommendations) are defined with the understanding that, unlike the previous model, they apply to absorption only. In addition, allowance is made for the use of absorption parameters for specific compounds. Specific tissues with associated detriment weighting are defined. Where the previous respiratory tract model was a single target organ, this model includes the thoracic lung, which replaces the previous target organ, and the extrathoracic lung, which is part of the remainder.

ICRP Publications 56, 67, 69, 70, and 72 contain metabolic models with age-specific parameters for various radioelements. Most of these models are of the same format as those provided in ICRP Publication 30—a transfer compartment out of which material is translocated to a number of organs and tissues. While a few of these have updated rate constants and gastrointestinal absorption parameters, most are the same as given in ICRP Publication 30. Several models include recycling material into organs and tissues and back into the bloodstream to be reabsorbed by other (or the same) organs and tissues. These can be quite complicated with up to 18 recycling compartments.

This set of models still relies on the GI tract model presented in ICRP Publication 30 and the physiological data presented in ICRP Publication 23. The only changes to the physiological data are breathing rates, volumes, and other parameters from the ICRP Publication 66 respiratory tract model. This publication has been recently updated in ICRP Publication 89, but this information has not yet been incorporated into the current system (ICRP 2003).

DERIVATION OF DOSE COEFFICIENTS

Dose coefficients are obtained by assuming some reference intake and calculating the committed effective dose from that intake based on the system of models that is in use. Committed effective dose in Sv is calculated

Table 1. Comparison of ICRP Publications 26 and 30 quantities, units, and terms.

ICRP 26 Quantity	ICRP 60 Quantity
Quality factor (Q)	Radiation weighting factor (w_r)
Weighting factor (w_T)	Tissue weighting factor (w_T)
Dose equivalent (H)	Equivalent dose (H_T)
Effective dose equivalent ($\sum w_T H_T$)	Effective dose (E)
Committed dose equivalent (H_{50})	Committed equivalent dose [$H_T(\tau)$]
Committed effective dose equivalent ($\sum w_T H_{50,T}$)	Committed effective dose [$E(\tau)$]
Stochastic effect	Stochastic effect (no change)
Non-stochastic effect	Deterministic effect

using eqns (2) and (3) where the SEE factor is in units of MeV g⁻¹ per transformation. The number of transformations in a single compartment, assuming no radioactive progeny, is obtained using the following equation:

$$U = N(0) \frac{\lambda}{k} [1 - e^{-k(50 \text{ y})}], \quad (5)$$

where:

- $N(0)$ = initial number of atoms at time $t = 0$;
- λ = radiological decay constant; and
- k = total removal rate constant.

The term following $N(0)$ in eqn (5) can be used as a *replacement function* to replace the exponential terms in a general kinetics equation:

$$N(t) = \sum_{i=1}^n a_i e^{-\gamma_i t}, \quad (6)$$

where γ_i = eigenvalue rate constant for the i^{th} term, and a_i = coefficient for the i^{th} term.

The use of γ rather than k implies an eigenvalue that is a solution to the general model including recycling between compartments and would replace the k in eqn (5).

Using matrices to obtain eigenvalues and eigenvectors

Pathway modeling using catenary kinetics, a common and extremely useful mathematical technique in health physics, usually requires solution of a number of differential equations describing movement of material between physical or mathematical compartments (Polig 2001; Potter 2002). For instance, a simple two-compartment model can be defined:

$$\begin{aligned} \frac{dN_1(t)}{dt} &= k_{2,1}N_2(t) - k_1N_1(t) \\ \frac{dN_2(t)}{dt} &= k_{1,2}N_1(t) - k_2N_2(t), \end{aligned} \quad (7)$$

where:

- $N_i(t)$ = functions describing the numbers of atoms in compartments i ;
- $k_{i,j}$ = rate constants describing movement of material along the singular pathway from compartment i to compartment j (partial removal rate constants); and
- k_i = rate constants describing total physical and radiological removal from compartment i (total removal rate constants).

These equations can be written in matrix form as:

$$\begin{bmatrix} \frac{dN_1(t)}{dt} \\ \frac{dN_2(t)}{dt} \end{bmatrix} = \begin{bmatrix} -k_1 & k_{2,1} \\ k_{1,2} & -k_2 \end{bmatrix} \begin{bmatrix} N_1(t) \\ N_2(t) \end{bmatrix}, \quad (8)$$

or

$$\dot{\mathbf{N}} = \mathbf{kN}. \quad (9)$$

Note that the coefficient matrix \mathbf{k} contains as its elements rate constants that have their indices reversed from the normal convention. Transforming an array into a matrix of this form can be accomplished by defining rate constants as in eqn (8) and operating on the transpose of the resulting matrix.

Solutions of the system of equations can be obtained using *eigenvectors* and *eigenvalues*. Eigenvalues are scalar quantities γ that are solutions to the equation:

$$|\mathbf{k} - \gamma\mathbf{I}| = 0. \quad (10)$$

Eigenvectors are column matrices \mathbf{x} that are solutions to the equation:

$$(\mathbf{k} - \gamma\mathbf{I})\mathbf{x} = 0. \quad (11)$$

The number of eigenvalues and eigenvectors will be the same as the number of rows and columns in the square matrix \mathbf{k} , which also is the number of compartments in the model. Upon calculating the eigenvalues and eigenvectors, solutions of the system are in the following form:

$$\begin{bmatrix} N_1(t) \\ N_2(t) \end{bmatrix} = c_1 \begin{bmatrix} x_{1,1} \\ x_{2,1} \end{bmatrix} e^{\gamma_1 t} + c_2 \begin{bmatrix} x_{1,2} \\ x_{2,2} \end{bmatrix} e^{\gamma_2 t}. \quad (12)$$

The coefficients c_1 and c_2 are determined by applying the boundary conditions for $t = 0$, usually $N_1(t) = N_1(0)$ and $N_i(t) = 0$ ($i \neq 1$) and solving for c_1 and c_2 .

Once this system of equations is solved, the replacement function can be applied. The c_i and $x_{j,i}$ values in eqn (12) can be multiplied together for each i^{th} equation to obtain the a coefficients in eqn (6). At this point, the replacement function derived from eqn (5) is applied and each i^{th} equation takes the form:

$$U_i = N(0) \sum_{j=1}^n \frac{a_{i,j}}{\gamma_{i,j}} [1 - e^{-\gamma_j(50 \text{ yr})}]. \quad (13)$$

This is the solution for the general case. Very often, because of the speed of biological processes, a short radioactive decay constant, or both, exponential terms in eqn (13) are very small ($50 \gg 1/\lambda$). Eqn (13) then reduces to:

$$U_i = N(0) \sum_{j=1}^n \frac{a_j}{\gamma_j}. \quad (14)$$

Note that, in the general case, each equation in the system has as many terms as there are compartments in the whole system. If there is no recycling within the system of compartments, the equations are further simplified, with the assumption used in eqn (14). First, each chain can be considered separately. This means that a single compartment can be a member of several chains and, therefore, have several equations describing the contribution from each chain. Within a chain, the equation takes the form of a product of *branching ratios*. These are ratios of rate constants with a partial removal rate constant describing the pathway in the numerator and the total removal rate constant for the associated compartment in the denominator. For example, an equation describing the number of disintegrations in the third compartment of a chain would be written:

$$U_3 = N(0) \frac{k_{1,2}}{k_1} \frac{k_{2,3}}{k_2} \frac{\lambda}{k_3}. \quad (15)$$

Now that the number of transformations in each compartment has been derived, appropriate SEE factors have to be applied (eqn 3). Originally, SEE factors for most radiations were contained in the supplements to ICRP Publication 30. ICRP Publication 68 references SEECAL, a computer program developed at Oak Ridge National Laboratory, as a source for these factors. These can be calculated from eqn (4) if the absorbed fractions are available. In general, absorbed fractions for alpha and beta particles of any energy are unity, unless the source organ is the bone. Bone-specific absorbed fractions were originally published in ICRP Publication 30. Absorbed fractions for photons were originally published in ICRP Publication 23. All of these absorbed fractions were updated in SEECAL (Christy and Eckerman 1993).

Numbers of disintegrations for each organ and tissue and SEE factors can be used with eqn (3) to calculate committed equivalent dose to each target organ and tissue. Committed effective dose is calculated using eqn (2) and the tissue weighting factors found in Table 2 (ICRP 1991a). If the initially assumed activity was some value such as 1 Bq and the final dose values were in Sv, the dose coefficients are in units of Sv/Bq.

DERIVATION OF INTAKE RETENTION FRACTIONS

Intake retention fractions are also necessary to calculate dose (Skrable et al. 1988; Potter 2002). An intake retention fraction is the fraction of an initial intake present in an organ, a tissue, or excreta at some time after intake. To fit bioassay data, an intake retention fraction or IRF represents the abscissa of the data set for which the data themselves are the ordinate. The IRF contains

Table 2. Organs and tissues included in the calculation of effective dose and associated tissue weighting factors.

Organ or tissue	Tissue weighting factor, w_r
Gonads	0.2
Bone marrow (red)	0.12
Colon	0.12
Lung	0.12
Stomach	0.12
Bladder	0.05
Breast	0.05
Liver	0.05
Esophagus	0.05
Thyroid	0.05
Skin	0.01
Bone surface	0.01
Remainder ^a	0.05

^a Remainder organs include muscle, brain, small intestine, kidneys, pancreas, spleen, thymus, uterus, adrenals, and extrathoracic airway.

the rate constants for the compartments in question, the time after intake, and a time increment for sample collection, if appropriate. IRFs can be derived for acute or chronic intakes.

Calculation of IRFs is very similar to the derivation of dose coefficients. The fraction of activity expected in any organ or tissue following an acute intake can be defined as:

$$r_i(t) = \frac{N_i(t)}{N_1(0)}, \quad (16)$$

where $N_i(t)$ = the number of atoms in the compartment of interest at time t after intake, and $N_1(0)$ = the initial number of atoms corresponding to the intake. Therefore, using eqn (16) to derive $r_i(t)$ is equivalent to solving the system of equations described by eqn (8) with the initial condition of $N_i(t) = 1$ at $t = 0$.

Rate constants are obtained from the models for each pathway in the respiratory tract model, gastrointestinal tract model, and metabolic model. Absorption and clearance rate constants between the models and excretion parameters are also obtained. These rate constants are defined as double-arrayed variables as described in the previous section. Total removal rate constants are represented also as double arrayed variables by using a repeated index (i.e., $k_{i,i}$). The system is solved for eigenvalues and eigenvectors after changing the sign of the diagonal rate constants and taking the transpose of the matrix defined by the double-arrayed variable. Coefficients are determined by applying boundary conditions and solving the resultant equations. Final equations for whole-body with and without the extrathoracic airways can be obtained by adding equations that define retention in the appropriate compartments.

To obtain final equations for incremental excretion, i.e., excretion over a finite time period beginning at some

time t after intake, a replacement function is again used. That replacement function is:

$$e^{-(\gamma+\lambda)}(1 - e^{\gamma\Delta t}), \quad (17)$$

where Δt = the incremental time period over which the sample is collected, usually assumed to be one day.

Once the intake retention functions are obtained, IRF values are calculated as appropriate for the bioassay measurement methods and times at which data were obtained.

DATA

Of course, no intake or dose estimate could be made without data of some kind. Bioassay data are commonly used in the calculation of intakes. Both in vivo and in vitro methods are used for bioassay. Air monitoring may also be used in some cases for intake estimation.

Bioassay

Lung counting is commonly used for radionuclides that are of a particular hazard to the lung tissue. These typically alpha-emitting and insoluble radionuclides include some gamma or x-ray component either from the radionuclide itself or from some progeny that is used for detection. Patients usually sit in a chair or lie on a bed while detectors operate. Historically, CsI/NaI or "phoswich" detectors have been used for lung counting, but newer systems use high-purity germanium (HPGe) or broad-energy germanium (BEGe) detectors, which provide much better resolution than phoswich detectors (Berger and Goans 1981). The HPS ANSI N13.30 *Performance Criteria for Radiobioassay* standard describes expectations for the ability of the method to detect certain levels of radioactivity in the lung (HPS 1996).

There are many different whole body counting systems. Individuals may lie down on a bed, stand up, or sit in a chair. Bed-type or stand-up counters scan either by passing detectors over the body or have stationary detectors positioned over particular organs and tissues. Whole-body counters that do not scan often exclude the head in the area of analysis. It is very important that this is understood by the internal dosimetrist. This is because at early times after an inhalation intake, much of the activity may be in the extrathoracic lung, and if this is not accounted correctly, the intake retention fraction used will be incorrect. As with lung counting, HPGe detectors are used most commonly in whole body counting systems.

Thyroid counting is used specifically for the detection of isotopes of iodine in the thyroid. The detector of choice for decades has been the NaI detector. Typically,

the patient either holds the detector to his or her throat or sits by a stationary detector with his or her throat against it.

The most common in vitro bioassay method is urinalysis. Depending on the biokinetics and the detection limits of the analysis method, either a 24-h urine sample or a single-void (spot) sample is obtained. These samples are either counted as excreted with minimal sample preparation (liquid scintillation, gamma spectrometry) or prepared radiochemically before analysis. Analytes of radiochemically prepared samples are usually electroplated onto a metal disc or filtered onto a filter paper. These samples are analyzed by alpha spectrometry or proportional counting. Feces and other biological material also can be analyzed in this way (Singh 1994).

While urine sampling is the more common practice, the use of fecal monitoring for insoluble radionuclides can greatly enhance an internal dosimetry program. Some care must be taken in the design and conduct of such a program. After an intake, some of the material deposited in the lung will be quickly cleared to the GI tract. If that material is insoluble, very little will be absorbed out of the small intestine, and, therefore, will be excreted in the feces. In this case, fecal monitoring can be a very sensitive indicator of intake for suspected exposure, but analysis of the data can be difficult due to the variability of the digestive tract. For routine measurements of insoluble compounds, the intake retention fraction for fecal measurements can be one or two orders of magnitude greater than that for urine. Therefore, fecal measurements can be much more sensitive. However, care must be taken to ensure that short-term clearance of the material from the respiratory tract is not occurring. This can be done by obtaining samples from individuals after a 1 or 2-wk vacation (NCRP 1986).

Air monitoring

The extent to which an internal dosimetry program is allowed to use air monitoring data for intake estimation varies based on the type and location (i.e., country) of the operation. There are different types of air monitoring, some of which are more suitable for evaluating personal exposure. Typically, fixed-head air monitoring of any type (e.g., alarming continuous air monitors) should not be used for the evaluation of personal exposure because the data obtained are typically not representative of the breathing environment. However, it may be considered in the determination of the intake scenario and physical or chemical properties, or used for dose calculation if no other data are available.

Breathing-zone air monitoring and personal air sampling are useful in maintaining an internal dosimetry program. Their particular uses include:

- estimation of low-level intakes for radionuclides where no method of analysis provides that ability (e.g., Pu);
- tracking/summation of exposures for comparison to routine monitoring requirements;
- timely determination of the magnitude of unexpected exposures; and
- timely feedback to operations personnel for the modification of workplace controls.

If radionuclides in use during the operation are known, samples can be counted by proportional counting and gross results obtained. Exposure can be determined using the ratio of the flow rate of the sampler and Reference Man's breathing rate as an intake retention fraction. Time of exposure is not needed.

Reference levels

The method of determination of the validity or meaning of specific bioassay results should be considered at the development of the internal dosimetry program. Reference levels are convenient parameters that can be used. They are determined by choosing some dose level at which a particular action will be performed and translating that dose level to a measurement value. This is done by applying the appropriate dose coefficient and IRF to the dose level. An example of a reference level is a "recording level," below which the measurement value will be recorded and above which some additional evaluation or sampling will occur. Another commonly used reference level is an "investigation level" where special monitoring is instituted. Recording and investigation levels should be developed for each bioassay type and frequency and actions to be taken written into procedures to ensure that internal dosimetry program objectives, both operational and regulatory, are met (ICRP 1997).

INTAKE CALCULATION

There is no standard method for intake calculation. Methods vary from simple to complicated, and many of the currently used methods require a specific computer application. Certainly, the effort used to calculate an internal dose should be in proportion to its magnitude—simple methods for low doses and more complicated for higher doses. It is important to understand, however, that different methods used result in sometimes greatly different intake values. In addition, there is no "right" answer to an internal dosimetry calculation. Because of the unknown magnitudes of errors in the biokinetics models as compared to the particular individual's metabolism and the sometimes

large random and systematic errors in bioassay methodology, it is difficult, if not impossible, to perform a determination of true intake and dose. What follows is a method that can be performed simply and gives a result that when visually compared to the data provides an acceptable answer (Skrable et al. 2002).

Fitting statistics

One way of determining a best fit to data is to choose a goodness-of-fit statistic and minimize it in some fashion. For instance, to perform a least-squares fit, the sum of the squares of the deviations between the data and the fit is usually minimized. Another statistic that can be minimized is the reduced chi-square (χ^2_v) statistic. Care must be taken when using this parameter, because the optimal value is unity rather than zero, and normalization should be performed to bring the value back to unity (Skrable et al. 2002).

The expected quantity of material in a bioassay compartment from any single intake can be written as:

$$\langle q_i \rangle = IF_i, \quad (18)$$

where $\langle q_i \rangle$ = expected bioassay measurement for a particular bioassay type and time after intake, I = intake, and F_i = appropriate intake retention fraction for the measurement.

The χ^2_v statistic is defined as:

$$\chi^2_v = \frac{1}{v} \sum_{i=1}^n \frac{(q_i - \langle q_i \rangle)^2}{\sigma_i^2}. \quad (19)$$

Substituting eqn (18) into eqn (19), reducing χ^2_v by taking the derivative with respect to each measurement, and solving for the intake, gives a general solution:

$$I = \frac{\sum_{i=1}^n \frac{q_i F_i}{\sigma_i^2}}{\sum_{i=1}^n \frac{F_i^2}{\sigma_i^2}}. \quad (20)$$

Selection of the appropriate variance model [$\sigma^2 = f(\langle q \rangle)$] will determine what type of fit is obtained.

A general variance model for data has been described of the form (Tries 2000):

$$\sigma_i^2 = \kappa \langle q_i \rangle + \phi^2 \langle q_i \rangle^2. \quad (21)$$

The significance of either term in eqn (21) depends on whether the greater source of variance is random or systematic. For bioassay data, where the largest variance is probably how representative the models are of actual

metabolism of an individual, the second term may be prevalent. Regardless, three types of fits will be shown: unweighted and two types of weighted fits relating to eqn (21).

When performing an unweighted fit, it is assumed that all variances are equal, that is that $\sigma_i^2 = v$. When substituting this into eqn (20), the variances divide out and the intake is given by:

$$I = \frac{\sum_{i=1}^n q_i F_i}{\sum_{i=1}^n F_i^2}. \quad (22)$$

To perform weighted fits, eqn (21) is used with eqn (20). If the first term in eqn (21) dominates, substituting into eqn (20) results in:

$$I = \frac{\sum_{i=1}^n q_i}{\sum_{i=1}^n F_i}. \quad (23)$$

This is also known as a “ratio of the means” fit since dividing the numerator and denominator by the number of data would, in fact, be the ratio of the means of q and F .

A weighted fit where the second term in eqn (20) dominates results in:

$$I = \frac{1}{n} \sum_{i=1}^n \frac{q_i}{F_i}. \quad (24)$$

This fit is called an “average of the slopes” since the equation does, in fact, calculate a slope at each point, sum those slopes, and divide by the number of points.

Note that the intake values represented by eqns (22), (23), and (24) are simple to calculate and the mathematics can be performed in a spreadsheet or even by use of a hand calculator. The equation used is dependent only on the variance model that is most representative of the data. This may be most evident by obtaining intakes followed by expectation values from each fit, graphing the expectation values with the data, and seeing which graph is most representative of the data. The average of the slopes method weights the later data more with respect to the earlier data. Because the models tend to be more representative during longer times post intake, it has been the author’s experience that this type of fit is usually more representative. It must be noted that the χ^2_v

statistic can no longer be used as a goodness-of-fit statistic, and some other test, such as a “runs” test, should be used for that purpose.

Error analysis

Errors in intake values obtained by the use of eqns (22), (23), and (24) can be obtained by propagating the errors in those equations. The general equation for the variance in I is given by:

$$\sigma_I^2 = \left[\sum_{i=1}^n \frac{F_i^2}{\sigma_i^2} \right]^{-1}. \quad (25)$$

Substituting the variances as was done above gives the following results:

$$\sigma_I^2 = \frac{v}{\sum_{i=1}^n F_i} \quad (26)$$

in the case of the unweighted fit,

$$\sigma_I^2 = \frac{\kappa I}{\sum_{i=1}^n F_i} \quad (27)$$

for the ratio of the means, and

$$\sigma_I^2 = \frac{\phi^2 I^2}{n} \quad (28)$$

for the average of the slopes. The parameters v , κ , and ϕ^2 are obtained by normalizing χ^2_v to unity:

$$v = \frac{1}{v} \sum_{i=1}^n (y_i - IF_i)^2 \quad (29)$$

for the unweighted fit (note that this is simply the standard error in q),

$$\kappa = \frac{1}{v} \sum_{i=1}^n \frac{(y_i - IF_i)^2}{IF_i} \quad (30)$$

for the average of the means (note that this is simply the best estimate of the χ^2_v statistic if $\kappa = 1$), and

$$\phi^2 = \frac{1}{v} \sum_{i=1}^n \frac{(y_i - IF_i)^2}{I^2 F_i^2} \quad (31)$$

for the average of the slope (note that this is also the best estimate of χ^2_v).

Calculation of error in intake is often overlooked because it is typically not required to be reported.

However, it is extremely important to show the reliability of the calculated intake with respect to the biological models, variance model, and random error.

Dose estimation

Once an intake with associated error has been calculated, it is very simple to infer a committed effective dose with associated error. ICRP Publication 68 contains dose coefficients for committed effective dose. Additionally, the ICRP has made available dose coefficients for committed equivalent dose to all of the organs and tissues listed in Table 2 in a database (ICRP 2001). Calculating dose is as simple as applying the dose coefficient to the intake value. This operation can be repeated with the intake error to obtain a value for the error in dose.

INTERNAL DOSIMETRY PROGRAM DESIGN

The design and implementation of an internal dosimetry program is usually overshadowed by the taking of data and calculating of dose. However, a properly designed program not only fulfills requirements, but also contributes to actual worker protection by providing quality control for the workplace control program. ICRP Publication 78 describes the following monitoring programs (ICRP 1997):

- routine monitoring—periodic measurements to meet requirements or test workplace controls;
- special or task-related monitoring—monitoring for a specific operation or as a response to an off-normal event; and
- confirmatory monitoring—monitoring of a particular population for quality control/assurance purposes.

An internal dosimetry program will usually include all of these types of monitoring.

To properly define and implement an internal dosimetry program, a number of questions should be answered. These questions are detailed in the following sections.

When is an internal dosimetry program required?

Requirements for internal dosimetry programs are contained in regulations or standards, depending on the overseeing body. For instance, in the United States, the Nuclear Regulatory Commission requires an internal dosimetry program if adults are “likely to receive” 10% of the ALI obtained from ICRP Publication 30. The Department of Energy requires a program for radiological workers “likely to receive” a committed effective dose equivalent of 0.001 Sv. ICRP Publication 78 recommends, “The results of workplace monitoring should give an indication of the likelihood of doses from intakes

exceeding 5 mSv a year.” In these cases, some analysis of material in process and work practices needs to be performed to show if a program is required (NRC 2004; U.S. DOE 2004; ICRP 1997).

Is my facility required to have an internal dosimetry program?

Now that the requirements have been determined, how can it be known if a facility meets the “likelihood” requirements? This question is easier to answer in a production-type facility than it is in a research facility. For a production facility, a number of factors come into play. For instance, has previous monitoring shown that the likelihood exists? One can use risk factors or resuspension factors also to make an estimate of the likelihood. For instance, a fraction 10^{-6} of radioactive material placed in a process in routine operations may enter the body of a worker over 1 y. The fraction 10^{-6} m^2/m^3 is also useful as a resuspension factor for surface contamination. When performing this analysis, all potential sources of internal contamination should be considered as well as work practices, occupancy factors, etc. (Brodsky 1980).

Is there some other reason to have an internal dosimetry program?

The facility may want to have an internal dosimetry program due to worker protection considerations. ICRP Publication 78 recommends:

“Experience has shown that workers involved in the following operations would normally require individual monitoring:

- 1 handling large quantities of gaseous and volatile materials, e.g., tritium and its compounds in large scale production processes, in heavy water reactors, and in luminescent product manufacture;
- 2 uranium mining and processing and fabrication of uranium and mixed oxide fuels;
- 3 processing of plutonium and other transuranic elements;
- 4 processing and use of thorium; and
- 5 production of large quantities of radionuclides and pharmaceuticals.”

Any operation that generates airborne contamination or has significant surface contamination such that an ingestion intake is possible is a candidate for an internal dosimetry program (ICRP 1997).

What are the monitoring objectives?

In the introduction to this paper, three monitoring objectives are described. These are:

- to provide timely feedback on workplace control;

- to initiate a medical intervention program; and
- to show compliance to a standard or regulation.

ICRP Publication 78 also provides three objectives:

- “1 to obtain an assessment of the committed effective dose and, where appropriate, the committed equivalent dose in significantly exposed tissues, so as to demonstrate compliance with managerial and regulatory requirements;
- 2 to contribute to the control of operation and the design of facilities; and
- 3 in the case of accidental exposure, to provide valuable information for the initiation and support of any appropriate health surveillance and treatment (ICRP 1997).”

These objectives are very similar to the bulleted objectives.

It is the opinion of this author that worker protection should be the primary objective of any part of a radiation protection program and that monitoring should be designed to fit that objective (first bulleted item and objective 2 above). Therefore, timeliness of feedback is an important consideration when designing an internal dosimetry program. Timeliness of feedback not only includes turnaround time for analysis, but also includes time between potential exposure and measurement. For this reason, air monitoring, including personal air monitoring, needs to be considered as part of the program.

What capabilities exist, are available, and/or are appropriate?

Once the need for an internal dosimetry program is determined, bioassay methods need to be determined. Considerations for type of bioassay include:

- method detection limits;
- frequency of measurement;
- type and timeliness of analysis method;
- ability to meet monitoring objectives;
- minimization of minimum detectable dose; and
- cost of measurement/analysis.

What do the measurements mean?

As described previously, reference levels should be derived prior to the institution of a radiobioassay program to ensure that the proper actions are taken and that the objectives of the internal dosimetry program are met.

FINAL THOUGHTS

In the first issue of *Health Physics*, K. Z. Morgan, the first President of the Health Physics Society and first Chairman of Committees 2 of the ICRP and NCRP, made

the following recommendations on the eve of the publication of ICRP Publication 2 (Morgan 1958):

- 1 Selection of MPC values for single exposure;
- 2 Selection of MPC values for human tracer studies;
- 3 Selection of MPC values based on gonad exposure;
- 4 Consider influence of secondary factors; and
- 5 Chronic biological studies of radionuclides.

Rather than respond to these recommendations, the internal dosimetry field took off in a different direction, choosing to evaluate individual intakes rather than track exposure.

While this author would not presume to second-guess Morgan, the following recommendations are offered for consideration:

- 1 A thorough evaluation of newer more complicated models with comparison to the simpler models using real human data, if possible, should be performed to gain an understanding of the usefulness of continuously updating biokinetic models;
- 2 Some method of measurement of actual content of radionuclides in individual organs and tissues needs to be developed. This would assist in the formation of new models and allow for determination of actual dose to individual organs for effective dose calculations;
- 3 A systematic approach to modifying model parameters to account for the difference in an individual's metabolism vs. Reference Man needs to be developed;
- 4 A better understanding of the accuracy of exposure and dose calculation and air sampling vs. bioassay data needs to be developed and methods obtained to allow for better exposure tracking and dose calculation, especially where there is a technology shortfall for bioassay measurements; and
- 5 Standard approaches to internal dose calculation should be developed. These approaches should be as simple as practicable so that an experienced health physicist can use them, and they should ensure that separate dose estimations from the same data do not differ by orders of magnitude.

The information presented in this article was just a small portion of all of internal dosimetry. I hope that the reader finds it interesting and thought provoking, and I thank *Health Physics* for the opportunity to prepare this work.

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REFERENCES

- Berger CD, Goans RE. A comparison of the NaI-CsI phoswich and a hyperpure germanium array for in vivo detection of the actinides. *Health Phys* 40:535–542; 1981.
- Blum T. Osteomyelitis of the mandible and maxilla. *Mon Labor Rev* 28:1222; 1929.
- Brodsky A. Resuspension factors and probabilities of intake of material in process (or “is 10^{-6} a magic number in health physics”). *Health Phys* 39:992–1000; 1980.
- Christy M, Eckerman KF. SEECAL: program to calculate age-dependent specific effective energies. Oak Ridge, TN: Oak Ridge National Laboratory; ORNL/TM-12351; 1993.
- Evans RD. Radium poisoning II: The quantitative determination of the radium content and radium elimination rate of living persons. *Am J Roentgenol Radium Ther Nucl Med* 37:368–378; 1937.
- Evans RD. Inception of standards for internal emitters, radon, and radium. *Health Phys* 41:437–448; 1981.
- Health Physics Society. Performance criteria for radiobioassay. McLean, VA: Health Physics Society; American National Standard HPS N13.30-1996; 1996.
- International Commission on Radiation Units and Measurements. Quantities and units in radiation protection dosimetry. Bethesda, MD: International Commission on Radiation Units and Measurements; ICRU Report 51; 1993.
- International Commission on Radiological Protection. Report of committee II on permissible dose for internal radiation. Oxford: Pergamon Press; ICRP Publication 2; 1959.
- International Commission on Radiological Protection. Report of the task group on reference man. Oxford: Pergamon Press; ICRP Publication 23; 1975.
- International Commission on Radiological Protection. Recommendations of the ICRP. Oxford: Pergamon Press; ICRP Publication 26; 1977.
- International Commission on Radiological Protection. Limits for intakes of radionuclides by workers. Oxford: Pergamon Press; ICRP Publication 30 Part 1; Ann. ICRP 2(3/4); 1979.
- International Commission on Radiological Protection. A compilation of the major concepts and quantities in use by ICRP. Oxford: Pergamon Press; ICRP Publication 42; Ann. ICRP 14(4); 1984.
- International Commission on Radiological Protection. Age-dependent doses to members of the public from intake of radionuclides: Part 1. Oxford: Pergamon Press; ICRP Publication 56; Ann. ICRP 20(2); 1989.
- International Commission on Radiological Protection. 1990 recommendations of the International Commission on Radiological Protection. Oxford: Pergamon Press; ICRP Publication 60; Ann. ICRP 21(1–3); 1991a.
- International Commission on Radiological Protection. Annual limits on intake of radionuclides by workers based on the 1990 recommendations. Oxford: Pergamon Press; ICRP Publication 61; Ann. ICRP 21(4); 1991b.
- International Commission on Radiological Protection. Age-dependent doses to members of the public from intake of radionuclides: Part 2 ingestion dose coefficients. Oxford: Pergamon Press; ICRP Publication 67; Ann. ICRP 23(3/4); 1993.
- International Commission on Radiological Protection. Human respiratory tract model for radiological protection. Oxford: Pergamon Press; ICRP Publication 66; Ann. ICRP 24(1–3); 1994.
- International Commission on Radiological Protection. Dose coefficients for intakes of radionuclides by workers, replacement of ICRP Publication 61. Oxford: Pergamon Press; ICRP Publication 68; Ann. ICRP 24(4); 1995a.
- International Commission on Radiological Protection. Age-dependent doses to members of the public from intake of radionuclides: Part 3 ingestion dose coefficients. Oxford: Pergamon Press; ICRP Publication 69; Ann. ICRP 26(1); 1995b.
- International Commission on Radiological Protection. Basic anatomical and physiological data for use in radiological protection: The skeleton. Oxford: Pergamon Press; ICRP Publication 70; Ann. ICRP 25(2); 1995c.
- International Commission on Radiological Protection. Age-dependent doses to members of the public from intake of radionuclides: Part 5 compilation of ingestion and inhalation dose coefficients. Oxford: Pergamon Press; ICRP Publication 72; Ann. ICRP 26(1); 1996.
- International Commission on Radiological Protection. Individual monitoring for internal exposure of workers, replacement of ICRP Publication 54. Oxford: Pergamon Press; ICRP Publication 78; Ann. ICRP 27(3/4); 1997.
- International Commission on Radiological Protection. The ICRP database of dose coefficients: workers and members of the public [CD-ROM]. Stockholm: International Commission on Radiological Protection; Version 2.0.1; 2001.
- International Commission on Radiological Protection. Basic anatomical and physiological data for use in radiological protection: reference values. Oxford: Pergamon Press; ICRP Publication 89; Ann. ICRP 32(3/4); 2003.
- Langam WH, Healy JW. Maximum permissible body burdens and concentrations of plutonium: Biological basis and history of development. In: Hodge HC, Stannard JN, Hursh JB, eds. *Handbook of experimental pharmacology, new series, vol 36*. New York: Springer-Verlag; 1978:569–592.
- Morgan KZ. Current status of the internal dose problem. *Health Phys* 1:125–134; 1958.
- National Bureau of Standards. Safe handling of radioactive luminous compound. Washington, DC: U.S. Government Printing Office; NBS Handbook H-27; 1941.
- National Council on Radiation Protection and Measurements. Use of bioassay procedures for assessment of internal radionuclide deposition. Bethesda, MD: National Council on Radiation Protection and Measurements; NCRP Report No. 87; 1986.
- Nuclear Regulatory Commission. Standards for protection against radiation. Washington, DC: U.S. Government Printing Office; 10CFR20; 2004.
- Polig E. Modeling the distribution and dosimetry of internal emitters: a review of mathematical procedures using matrix methods. *Health Phys* 81:492–501; 2001.
- Potter CA. Intake retention fractions developed from models used in the determination of dose coefficients developed for ICRP publication 68—Particulate inhalation. *Health Phys* 83:594–789; 2002.
- Singh N. Analytical measurements for in vitro bioassay of radionuclides. In: Raabe OG, ed. *Internal radiation dosimetry*. Madison, WI: Medical Physics Publishing; 1994: 409–430.
- Skrable KW, Chabot GE, French CS, La Bone TR. Intake retention functions and their applications to bioassay and the estimation of internal radiation doses. *Health Phys* 55:933–950; 1988.
- Skrable KW, French CS, Chabot GE, Tries M, La Bone TR. Variance models for estimating intakes from repetitive

- bioassay measurements. In: Bolch WE, ed. Practical applications of internal dosimetry. Madison, WI: Medical Physics Publishing; 2002: 257–305.
- Spoor NL, Hursh JB. Protection criteria. In: Hodge HC, Stannard JN, Hursh JB, eds. Handbook of experimental pharmacology, new series, vol 36. New York: Springer-Verlag; 1978: 241–270.
- Stannard JN. Radioactivity and health, a history. Richland, WA: Pacific Northwest Laboratory; DOE/RL/01830-T59; 1988.
- Tries MA. Applications of a quadratic variance model for counting data. *Health Phys* 78:322–328; 2000.
- U.S. Department of Energy. Occupational radiation protection. Washington, DC: U.S. Government Printing Office; 10CFR835; 2004.
- OTHER REFERENCES NOT CITED**
- In the 50 years of publication (excluding recent months) of *Health Physics*, there have been some 916 papers on topics in internal dosimetry. This includes 52 papers on standards and limits, 64 on calculation of internal dose, 84 on general models, 141 on bioassay techniques and issues, 91 on enhancing elimination of radionuclides, and a whopping 484 papers on metabolism of specific radionuclides. While one certainly cannot read them all, I have listed a sampling below that I have found interesting and/or useful in addition to those listed above.
- Brodsky A. Exact calculation of probabilities of false positives and false negatives for low background counting. *Health Phys* 63:198–204; 1992.
- Cheng YS, Dahl AR, Jow HN. Dissolution of metal tritides in a simulated lung fluid. *Health Phys* 73:633–638; 1997.
- Dolphin GW, Eve IS. Dosimetry of the gastrointestinal tract. *Health Phys* 12:163–172; 1966.
- Eakins JD, Hutchinson WP, Lally AE. The radiological hazard from tritium sorbed on metal surfaces. *Health Phys* 28:213–224; 1975.
- Eidson AF, Griffith Jr. WC. Techniques for yellowcake dissolution studies in vitro and their use in bioassay interpretation. *Health Phys* 46:151–163; 1984.
- Eidson AF, Mewhinney JA. In vitro dissolution of respirable aerosols of industrial uranium and plutonium mixed-oxide nuclear fuels. *Health Phys* 45:1023–1037; 1983.
- Evans RD. Remarks on the maximum permissible deposition of plutonium in man, and the safety factors in the pivot point radiation protection guide of 0.1 μC of radium in man. *Health Phys* 8:751–752; 1962.
- Eve IS. A review of the physiology of the gastrointestinal tract in relation to radiation doses from radioactive materials. *Health Phys* 12:131–161; 1966.
- Healy JW. The ICRP dose limitation system—solution or problem? *Health Phys* 42:407–413; 1982.
- Johnson JR, Dunford DW. Dosimetric models of ^3H from skin absorption following contact with T2-contaminated surfaces. *Health Phys* 48:110–113; 1985.
- Kropf RF, Wang Y, Cheng YS. Self-absorption of tritium betas in metal tritide particles. *Health Phys* 75:398–404; 1998.
- Langham WH, Bassett SH, Harris PS, Carter RE. Distribution and excretion of plutonium administered intravenously to man. *Health Phys* 38:1031–1060; 1980.
- Turner JE, Wright HA, Hamm RN. A Monte Carlo primer for health physicists. *Health Phys* 48:717–733; 1985.
- Vennart J. Limits for intakes of radionuclides by workers: ICRP publication 30. *Health Phys* 40:477–484; 1981. ■ ■