2.0 General Methods for Internal Dosimetry

The HIDP uses the fundamental concepts described by the ICRP for calculations of intake, deposition, and dose. The basic concepts and techniques are those described in ICRP 30, Part 1 and its supplement (1979a and b), including specific effective energy (SEE) factors, Reference Man parameters, annual limit on intake (ALI), and derived air concentration (DAC), with radiation quality factors and organ/tissue weighting factors as mandated by 10 CFR 835. The biokinetic models used to describe distribution, retention, and excretion for various radionuclides are described in the pertinent chapters of this manual. Generally, preference is given to well-documented and peer-reviewed models; particularly those published by the ICRP. This chapter summarizes the calculational methods and factors most commonly used for intake assessment and internal dosimetry. In some cases, discussion is included concerning scientific recommendations that are currently incompatible with regulatory requirements. Such discussion is intended to provide guidance for alternate assessments appropriate for purposes other than regulatory compliance. This chapter also describes the various reference levels and derived reference levels used by the HIDP.

2.1 Radiation Quality and Tissue Weighting Factors

The quality factors and tissue weighting factors of 10 CFR 835 are used for routine calculations. Generally, these factors have been incorporated into the computer codes and dose coefficients used in the calculations. These factors are consistent with those also found in 10 CFR 20 and ICRP 30 (1979a). Some significant differences in tissue weighting factors are found in the recommendations of ICRP 60 (1990), however these newer values are not consistent with 10 CFR 835, and thus are not used for compliance dose calculations. Table 2.1 provides a comparison of the radiation quality factors pertinent to the HIDP, and Table 2.2 compares the tissue weighting factors.

Table 2.1. Comparison of Radiation Quality Factors

Radiation	10 CFR 835	10 CFR 20	ICRP 30	ICRP 60
Alpha	20	20	20	20
Proton	10	10	10	5
Beta	1	1	1	1
Gamma, X	1	1	1	1

Table 2.2. Comparison of Organ/Tissue Weighting Factors

Tissue or Organ	10 CFR 835	10 CFR 20	ICRP 30	ICRP 60
Gonads	0.25	0.25	0.25	0.20
Breasts	0.15	0.15	0.15	0.05
Red Bone Marrow	0.12	0.12	0.12	0.12
Lungs	0.12	0.12	0.12	0.12
Thyroid	0.03	0.03	0.03	0.05
Bone Surfaces	0.03	0.03	0.03	0.01
Colon	-	-	-	0.12
Stomach	-	-	-	0.12
Bladder	-	-	-	0.05
Liver	-	-	-	0.05
Esophagus	-	-	-	0.05
Skin	-	-	-	0.01
Remainder	0.06 for each of 5 other organs with highest dose	0.06 for each of 5 other organs with highest dose	0.06 for each of 5 other organs with highest dose	0.05 total, with maximum of 0.025 to any single tissue

2.2 Biokinetic Models

Biokinetic models are used to describe the deposition and movement of material throughout the body. The ICRP 30 models for the respiratory tract, gastrointestinal (GI) tract, and metabolic distribution are used for most assessments (ICRP 1979a, b; 1980; 1981a, b; 1982a, b; 1988b). Specific metabolic retention and excretion models for various elements are described in the corresponding chapters of this manual. If adjustments are made to the parameter values of these models, those adjustments are explained in the documentation associated with the assessment.

Intake retention functions (IRFs) combine various biokinetic models to provide an expression of the amount of a radionuclide retained in a compartment of the body (or excreted by a particular pathway) as a fractional value of the amount of the intake. Values of the functions at various times post intake are tabulated in this manual for intake circumstances of greatest interest to the HIDP. Other common sources of IRF values that may be used include the computer code CINDY (code for internal dosimetry; Strenge et al. 1992), ICRP 54 (1988a), NUREG 4884 (Lessard et al. 1987), and peer-reviewed literature or hand calculations.

The CINDY computer code is the preferred code used for internal dosimetry by the HIDP. This code incorporates the ICRP 30 lung and GI tract models, along with the metabolic distribution models to give bioassay projections, intake assessments based on bioassay data, and estimates of dose equivalent.

Committed dose coefficients, $h_{T,50}$ and $h_{E,50}$, sometimes referred to as dose conversion factors, are the factors that express the committed tissue or effective dose equivalent, respectively, for a unit intake. They are derived based on a specified set of conditions. The dose coefficients tabulated in this manual are for circumstances most commonly encountered at Hanford. Typically, these are transportable injection (instant uptake) and inhalation of class D, W, or Y materials, assuming a 1-:m or 5-μm activity median aerodynamic diameter (AMAD) particle size. Other tabulations of dose coefficients that might be useful include Federal Guidance Report No. 11 (EPA 1988), ICRP 30 supplements, and the newer ICRP publications, as summarized in publications 68, 72, and 78 (ICRP 1994b; 1996; 1997). In addition, coefficients for differing conditions can be calculated using the CINDY computer code.

2.3 Bioassay Measurements

Bioassay is defined as the direct measurement of radioactivity in the body or the indirect measurement of radioactivity in the body by analyzing material excreted or otherwise removed from the body. Direct measurements are commonly called in vivo measurements, and use detector systems such as whole body counters, lung counters, and wound counters. Indirect measurements are called in vitro measurements, and involve the laboratory analysis of material excreted or removed from the body. In vitro measurements may include urine, feces, tissue samples, blood, or sputum. As a practical matter, most in vitro bioassays are made using urine or feces, and these measurements are generically referred to at Hanford as excreta bioassay.

2.4 Internal Dosimetry Assessments

The HIDP uses intake assessment as the principal means for most dose evaluation, with internal doses calculated based on estimated intake. The intake is estimated using available data, preferably bioassay measurements, but exposure time to air concentrations may also be used. The 50-year committed effective dose equivalent (CEDE) and any appropriate 50-year committed organ or tissue dose equivalents are calculated based on the intake. In some cases (notably, tritium), dose may be directly calculated from bioassay

measurements, with intake subsequently estimated based on the assessed dose. The 50-year committed dose equivalents, assigned to the year of intake, are used as the basis for compliance monitoring.

The concept of deposition assessment was often used by the HIDP through 1993, particularly for the assessment of plutonium, and is described here primarily for historical reference to archived evaluations. Rather than calculating an intake, this method used bioassay data to estimate the amount of material initially deposited in the intake compartments of interest. For example, instead of calculating a total inhalation intake, the bioassay data would be used to estimate the quantity initially retained in long-term compartments of the lung. In addition, a "presystemic deposition" was estimated to be the amount initially deposited in a compartment that would eventually translocate to the systemic circulation. (The term "presystemic deposition" was coined for use in the *Technical Basis for Internal Dosimetry at Hanford* [Sula et al. 1989]. Prior to that document the term "deposition" had been used, often very imprecisely.) Clearance rates were estimated for both non-systemic and presystemic depositions. Compliance with regulatory requirements was demonstrated either by calculating annual (not committed) dose equivalents to critical organs for comparison with the radiation protection standards in effect at the time, or, prior to 1989, by comparing the presystemic deposition with the maximum permissible body burden (MPBB). Tabulations of MPBBs in ICRP 2 (1959) or earlier National Bureau of Standards (NBS) handbooks (NBS 1953; 1959) were used as the radiation protection standards. In 1989, the ICRP 26 (1977) system was implemented for calculating effective dose equivalent using tissue weighting factors applied to organ and tissue dose equivalents, and calculated annual doses (not committed doses) were used as the basis for compliance. With the 1994 implementation of the DOE *Radiological Control Manual* (DOE 1994), compliance monitoring became based on assigning the committed dose equivalent to the year of intake.

2.5 Organs or Tissues of Concern

The DOE has established limiting values for occupational exposure to radiation in 10 CFR 835.202. These values include a limit on dose to individual organs or tissues to prevent deleterious nonstochastic effects, and a limit on the effective dose equivalent based on the risk of stochastic effects. Requirements for recording committed dose equivalents to organs and tissues of concern as well as the committed effective dose equivalent are given in 10 CFR 835.702(c)(4). However, neither the rule nor its implementation guide (DOE 1999) specifically defines "organs and tissues of concern." Practices for recording doses to "organs and tissues of concern" are defined in the

Hanford Internal Dosimetry Program Manual (PNNL-MA-552).[a] In cases involving relatively small effective dose equivalents, there may be no single organ that meets the recording criteria, whereas for a very significant exposure, several organs may qualify. Candidate organs and tissues used by the HIDP are those identified by ICRP 30 (1979a) and the tissue weighting factors of 10 CFR 835, however the element-specific chapters of this manual (PNNL-MA-860) narrow those candidates to the appropriate organs and tissues. As noted below, doses received by localized tissues are not included in either the assessment of effective dose equivalent or in the assessment of dose equivalent to organs and tissues of concern.

Intakes of radionuclides via wounds may result in the irradiation of local tissues at the wound site, as well as regional lymph nodes that drain the wound region. Because of their small mass, the absorbed dose to the regional lymph nodes may greatly exceed that to other tissues. Evidence from studies of experimental animals suggests that the lymph nodes are not primary sites for development of radiation-induced malignant disease (Nenot and Stather 1979). For this reason, there has been no attempt by either the ICRP (1979a) or the Biological Effects of Ionizing Radiation (BEIR) Committee (National Research Council 1988) to derive stochastic risk estimates for lymphatic tissue. Similarly, the irradiation of local tissues at the wound site is not considered to carry significant risk of carcinogenesis.

Concentrated activity in such localized sources can be expected to result in relatively high doses and cell death or tissue fibrosis (e.g., scar tissue) within a limited area, but unless this area comprises more than a minor fraction of the organ/tissue, there will likely be no observable nonstochastic effect at any dose. Assessment of organ or tissue dose equivalent from highly localized sources, made by averaging the energy deposited in the organ over the organ mass, is not a relevant measure for comparison with the limiting values for assessed dose based on nonstochastic effects. Furthermore, in most situations, it is not possible to determine the actual mass of affected tissue for computing the absorbed dose. Because the absorbed dose is highly nonuniform over the tissue and only a limited number of cells within the organ/tissue are affected, the use of dose equivalent for assessing this localized exposure is not valid.

(a) Pacific Northwest National Laboratory. 1997. *Hanford Internal Dosimetry Project Manual*. PNNL-MA-552, Richland, Washington. (Internal manual.) Available URL: http://www.pnl.gov/eshs/pub/pnnl552.html

For these situations, the HIDP will estimate the quantity of radionuclide(s) locally deposited and the projected retention half-time. These estimates become part of the individual's radiation protection record, but are not used for determining compliance with either the stochastic or nonstochastic limits. This approach is analogous to the approach described in 10 CFR 835.205(b)(3) for irradiation of limited areas of the skin.

2.6 Particle Size

The ICRP 30 (1979a) lung model recommended a default particle size of 1-μm AMAD, and that recommendation was used by the HIDP for most applications through the 1980s and 1990s. The ICRP 66 lung model (ICRP 1994a) provided two particle size recommendations for radiological protection purposes, in the absence of more specific information. For occupational exposure circumstances, a reference AMAD of 5 μm was recommended, and a 1-μm AMAD was recommended for exposures in the general environment.

Dorrian and Bailey (1995) reported on a survey of 52 publications addressing radioactive aerosol particle size distributions in the workplace. Reported values ranged from 0.12 μm to 25 μm, and were well fitted by a lognormal distribution with a median value of 4.4 μm. They noted that nuclear power and nuclear fuel handling facilities gave median values of about 4 μm. Uranium mills gave a median value of 6.8 μm, with AMADs frequently above 10 μm. High temperature and arc saw cutting operations generated submicron particles. They concluded that a 5-μm AMAD was a realistic default value for occupational exposure to unknown aerosols, and considered that value a better choice than the 1-μm value of ICRP 30. They also cautioned that, where possible, particle sizes should be measured for individual work practices to provide realistic parameters for dose assessment, because the 5-μm value of the ICRP 66 lung model was chosen to be deliberately realistic rather than conservative.

Kelso and Wraight (1996) reported on 50 AMAD measurements associated with reactor fuel reprocessing in several buildings at Sellafield. They found a mean value of 3.7-μm AMAD over the six buildings examined, with results consistently larger than the 1-μm value. They also concluded that the use of 5 μm as a default particle size was reasonable as a realistic rather than conservative assumption for occupational aerosols.

Heid and Jech (1972) reported that measurements in the majority of Hanford incidents indicated a mean particle size of 4-µm for plutonium oxide particles. Palmer, Perkins, and Stuart (1964) reported submicron particle sizes for radon in mines. These two reports provide some older historical data in support of the ICRP 66 lung model recommendations.

This technical basis work for the HIDP addresses both 1-µm and 5-µm AMAD particle sizes. This is done as a point of reference for the environmental and occupational radiation protection recommendations of ICRP 66, and as a cross-over to mark the change from the 1-µm default of ICRP 30 to the 5-µm default of ICRP 66 for occupational radiation protection. It is the intent of the HIDP to follow the particle size recommendations of ICRP 66 and assume occupational exposure to radioactive aerosols of a 5-µm particle size, unless exposure information suggests otherwise.

2.7 Assumed Date of Intake

The actual intake time or period, when that time is known, is used for assessment of intake and dose.

When the actual intake time or period is not known, it is necessary to identify the probable intake date(s). This may be done by considering available evidence, such as air monitoring results, contamination surveys, operating periods, and previous bioassay measurement results. After the intake time is narrowed to a probable time period, it is assumed that an acute intake occurred at the midpoint of that period. This approach is consistent with recommendations of the ICRP (1988a; 1997) and the National Council on Radiation Protection and Measurements (NCRP 1987).

If the evidence suggests that a chronic intake is more reasonable, it is assumed that the chronic intake occurred uniformly throughout the probable exposure period.

For describing the capability of a bioassay program (i.e., the minimum detectable intake or dose associated with a bioassay measurement protocol), the intake is assumed to be at a worst-case date (i.e., the minimum IRF value for the interval is used). Typically, that date is the longest elapsed time between measurements.

2.8 Intake Pattern

Occupational intakes in a well-engineered and -operated facility usually occur as acute inhalations due to unplanned or unanticipated events. Thus, acute inhalations are used for most bioassay program designs and as default intake patterns for assessment of high routine bioassay samples. Exceptions to this may include tritium exposures, which can be expected to occur as acute or chronic uptake events combining both inhalation and absorption. Historically at Hanford, there has also been planned chronic exposure to uncontained uranium in several facilities, however the work associated with such patterns is now quite rare. Very low-level chronic exposure, below the sensitivity of normal air sampling and bioassay monitoring may be present for areas of uncontained radioactivity.

2.9 Interpretation of Bioassay Program Capability

Bioassay program capability (i.e., sensitivity) is described by the minimum detectable intake (MDI) and its associated minimum detectable dose (MDD), based on an assumed bioassay measurement equal to the minimum detectable activity or amount (MDA) and an assumed time between measurements.

Occasionally it is desirable to make a statement based on an actual bioassay measurement, showing no detectable result, as to what might be the potentially undetected dose associated with the measurement. When a bioassay measurement has been made with a result showing no detection, the MDA value, rather than the critical level for detection (L_c) should be used as a basis for determining a potentially undetected dose. The MDA and critical level for detection concepts are discussed in Appendix B of this manual.

2.10 Normalizing Bioassay Data

Indirect bioassay data may be normalized differently based on the sampling protocol. Generally, Hanford urine data are automatically normalized to a total 24-hour excretion by use of the standard "simulated 24-hour" sampling protocol of collecting all urine voided between 30 minutes before retiring at night and 30 minutes after rising in the morning for two consecutive nights (NCRP 1987). This protocol was originated at Hanford in the mid-1940s, based on unpublished work by J. W. Healy.[(a)] Medley, Kathren, and Miller (1994) identified a potential bias of up to a factor of 2 for this protocol.

(a) Personal correspondence, J. W. Healy to E. H. Carbaugh.

If the sample is collected properly, a total or simulated 24-hour urine sample result is used as is; no further normalization is done. A proper 12-hour sample result is normalized by doubling the result. If it is suspected that a sample has not been provided according to instructions, several approaches are considered for normalization. A sample that is supposed to contain 24-hour excretion may not be analyzed if the volume is less than 500 ml because the volume is too small to represent a true 24-hour collection, and the worker may be asked to provide another sample. Alternatively, the result may be 1) normalized to 24-hour excretion based on information from the provider, 2) ignored, or 3) normalized by volume to 24-hour excretion. To normalize by volume, 1400 ml for males and 1000 ml for females (from Reference Man [ICRP 1974]) should be used for 24-hour excretion unless the person-specific daily excretion rate is known.

Normalization by creatinine or specific gravity has been suggested (NCRP 1987; NIOSH 1974; Anderson et al. 1995; Karpas et al. 1998; Duke 1998). However, various studies suggest that normalization by these methods does not provide any improved confidence in the result over normalization by time or volume (Jackson 1966; Kim 1995; Boeniger, Lowry, and Rosenberg 1993; and Graul and Stanley 1982). The best way to ascertain if the sample represents 24-hour excretion may be to simply ask the worker providing the sample. (Harris 2000)

The one exception to the above discussion concerns the analysis for tritium in urine. Because tritium is usually considered to be in equilibrium with body water, dosimetry can be accomplished using urine concentration rather than a daily excretion rate.

The fecal excretion for Reference Man (ICRP 1974) for adults ranges from 60 to 500 g/day, with a recommended average of 135 g/day for an adult male and 110 g/day for an adult female. Note that these values represent excretion “per day,” not excretion “per bowel movement.” When a single bowel movement is collected, it is generally interpreted as representing excretion for one day. If the sample is greater than 60 g, no normalization is used. If the sample is less than 60 g, normalizing to 135 g for males and 110 g for females may be appropriate.

If total accumulated fecal excretion over a time period was requested and there is no apparent reason to suspect that total excretion was not provided, then all sample results should be used as they are, without regard for the mass of individual samples. If excretions were missed during the time period, then normalization of the total mass to the

total mass expected based on the reference values given above should be used.

2.11 Fitting Bioassay Data to Biokinetic Models

The assessment of intakes or internal dose requires fitting bioassay data to an appropriate biokinetic model to assess the integrated retention function (i.e., the cumulative activity or the number of radionuclide transformations over the time period of interest). Normally the HIDP uses the CINDY computer code (Strenge et al. 1992) to make these assessments. Curve-fitting routines within the code are addressed in Appendix D. Although curve fitting for a given type of bioassay (e.g., a set of urine samples from one person) can be refined by rigorous mathematics, often excretion and retention curves for a single intake of an individual do not lead to compatible intake estimates. In such cases, the dosimetrist must exercise considerable judgment in estimating the true intake.

2.12 Reference Levels and Derived Reference Levels

A reference level is a predetermined value of a quantity that triggers a specified course of action when exceeded or expected to be exceeded. Reference levels at Hanford are expressed as dose-based or intake-based. Derived reference levels are the measurement values for particular bioassay or air sampling results that correspond to a more general reference level under specifically defined circumstances. The reference and derived reference levels used in this document are the following:

- screening level (SL)—The level below which a bioassay measurement need not be considered for investigation of intake and assignment of dose. The Hanford SL is based on a committed effective dose equivalent of 10 mrem.

- derived screening level (DSL)—The value of a bioassay measurement or airborne exposure estimate corresponding to a committed effective dose equivalent of 10 mrem for the referenced conditions or an estimated normal environmental levels.

- verification level (VL)—The level above which an attempt should be made to confirm the intake as real (i.e., special follow-up measurements should be made to a high routine measurement). The Hanford VL is 100-mrem committed effective dose equivalent.

- investigation level (IL)—The level above which a bioassay or air monitoring result shall be investigated, to the extent reasonable, to determine actual conditions and parameters for dose evaluation. An investigation may involve special measurements, work history review, determination of material form, and modification of biokinetic parameters. The Hanford IL is 100-mrem committed effective dose equivalent. In practice, Hanford does not discriminate between the VL and the IL.

- derived investigation level (DIL)—The bioassay measurement or airborne exposure measurement corresponding to a committed effective dose equivalent of 100 mrem for the referenced conditions.

- derived compliance level (DCL)—The bioassay measurement level corresponding to the 10 CFR 835 dose limit, i.e., 5-rem committed effective dose equivalent or 50-rem committed organ/tissue dose equivalent.

Values of DCLs, DILs, and DSLs are tabulated in the various chapters of this document.

2.13 Fundamental Relationships

The first principles equation for dose equivalent rate to an organ or tissue is described as

$$\dot{H}_T = 1.6\text{x}10^{-10} \times SEE(T \leftarrow S) \times A_s \qquad (2.1)$$

where $\dot{H}_T$ = dose equivalent rate in the target organ or tissue (T) from radioactive transformations in a source organ or tissue, in units of sieverts per second (Sv/s),

$SEE(T \leftarrow S)$ = specific effective energy deposited in the target organ or tissue from a radionuclide transformation in a source organ or tissue in units of MeV per gram – transformation (MeV/g-trans), and

A_s = radioactivity present in the source organ or tissue in becquerels (Bq).

Integrating Equation (2.1) with respect to retention time gives the following dose equivalent:

$$H_T = 1.6 \times 10^{-10} \times SEE(T \leftarrow S) \times A_s \int_0 R(t)dt \qquad (2.2)$$

where H_T = dose equivalent in sieverts (Sv),

$R(t)$ = retention function in the source organ or tissue, and

$A_s \int R(t)dt$ = total number of transformations in the source organ over the time interval of interest. This latter term is also known as cumulative activity. For most internal dosimetry calculations, the integral is solved for t = 50 years (or 1.58 x 10^9 seconds). ICRP 30 nomenclature identifies this term as U_s.

In conventional health physics units, Equation (2.2) is expressed as

$$H_T = 51.2 \times SEE(T \leftarrow S) \times A_s \int_0 R(t)dt \qquad (2.3)$$

where H_T = rems,
A_s = microcuries (µCi), and
t = the time interval in days.

A few fundamental relationships are repetitively used for most internal dosimetry calculations. Some of these relationships are described conceptually in the following equations. Because these are intended to be conceptual relationships, no units or unit conversion factors are shown. It is understood that consistency in units will be addressed by the specific application of the relationship.

Bioassay Result = Intake × Intake Retention (or Excretion) Fraction

$$M(t) = I \times IRF(t) \qquad (2.4)$$

Intake Estimate from a Single Bioassay Measurement

$$I = \frac{M(t)}{IRF(t)} \qquad (2.5)$$

Intake Estimate from Air Concentration Data

$$I = \frac{C_{air} \times \text{Breathing Rate} \times \text{Exposure Duration}}{\text{Respiratory Protection Factor}} \qquad (2.6)$$

Calculation of Airborne Exposure (DAC-hours)

$$\text{DAC} - \text{hours} = \sum_{i=1}^{n} \frac{C_{air,i}}{DAC_i} \times \text{Duration of Exposure (hours)} \qquad (2.7)$$

Annual Limit on Intake, the most limiting from Dose Limit and Dose Coefficient

$$\mathrm{ALI} = \frac{\text{Dose Limit}}{\text{Dose Coefficient}} = \frac{5\ \text{rem}}{h_{E,50}}, \text{ or } \frac{50\ \text{rem}}{h_{T,50}} \tag{2.8}$$

Intake Estimate from Airborne Exposure (DAC-hours) and Annual Limit on Intake

$$I = \frac{\mathrm{ALI}}{2000\ \mathrm{DAC-hours}} \times \mathrm{DAC-hours} \tag{2.9}$$

Committed Effective Dose Equivalent based on Intake and Dose Coefficient

$$H_{E,50} = I \times h_{E,50} \tag{2.10}$$

Committed Organ/Tissue Dose Equivalent based on Intake and Dose Coefficient

$$H_{T,50} = I \times h_{T,50} \tag{2.11}$$

Committed Dose from DAC-hours

$$H_{E,50} = \mathrm{DAC-hours} \times \frac{5\ \text{rem}}{2000\ \mathrm{DAC-hours}} \tag{2.12}$$

(for stochastically based DACs)

$$H_{T,50} = \mathrm{DAC-hours} \times \frac{50\ \text{rem}}{2000\ \mathrm{DAC-hours}} \tag{2.13}$$

(for deterministically based DACs)

Minimum Detectable Intake (MDI)

$$\mathrm{MDI} = \frac{\mathrm{MDA}}{\mathrm{IRF(t)}} \tag{2.14}$$

Minimum Detectable Dose (MDD)

$$\mathrm{MDD} = \mathrm{MDI} \times h_{E\ \mathrm{or}\ T,50} \tag{2.15}$$

2.14 References

10 CFR 20. 1999. U.S. Nuclear Regulatory Commission. "Standards for Protection Against Radiation." U.S. Code of Federal Regulations.

10 CFR 835. 1999. U.S. Department of Energy. "Occupational Radiation Protection." U.S. Code of Federal Regulations.

Anderson, J., B. Kahn, R. Rosson, E. Kim, and T. R. La Bone. 1995. "Use of creatinine measurement to confirm the time interval of a urine sample." *Rad. Prot. Management*, 12(5):51-65.

Boeniger, M. F., L. K. Lowry, and J. Rosenberg. 1993. "Interpretation of urine results used to assess chemical exposure with emphasis on creatinine adjustments: A review." *Am. Ind. Hyg. Assoc. J.* 54(10):615-627.

Duke, K. 1998. "Use of the urinary excretion of creatinine in plutonium in urine bioassay." *Rad. Prot. Dosim.* 79(1-4):125-128.

Dorrian, M. D., and M. R. Bailey. 1995. "Particle size distributions of radioactive aerosols measured in workplaces." *Rad. Prot. Dosim.* 60(2):119-133.

Graul, R. J. and R. L. Stanley. 1982. "Specific gravity adjustment of urine analysis results." *Am. Ind. Hyg. Assoc. J.* 43(11):863.

Harris, Shelley A., James T. Purdham, Raul N. Covey, and Andrea M. Sass-Kortask. 2000. "An evaluation of 24-hour urinary creatinine excretion for use in identification of incomplete urine collections and adjustment of absorbed dose of pesticides." *Am. Ind. Hyg. Assoc. J.*61:649-657.

Heid, K. R., and J. J. Jech. 1972. "Prompt handling of cases involving accidental exposure to plutonium." In *Health Physics Operational Monitoring*, Vol. 3, Proceedings of the Third Annual Health Physics Society Midyear Symposium, January, 29-31, 1969, C. A. Willis and J. S. Handloser, eds., Gordon and Breach Science Publishers, New York.

International Commission on Radiological Protection (ICRP). 1959. *Report of Committee II on permissible dose for internal radiation.* ICRP publication 2, Pergamon Press, New York.

International Commission on Radiological Protection (ICRP). 1974. *Report of the task group on Reference Man.* ICRP publication 23, Pergamom Press, New York.

International Commission on Radiological Protection (ICRP). 1977. "Recommendations of the International Commission on Radiological Protection." (ICRP publication 26.) *Annals of the ICRP*, 1:3, Pergamon Press, New York.

International Commission on Radiological Protection (ICRP). 1979a. "Limits for intakes of radionuclides by workers." (ICRP publication 30, part 1.) *Annals of the ICRP*, 2:3-4, Pergamon Press, New York.

International Commission on Radiological Protection (ICRP). 1979b. "Limits for intakes of radionuclides by workers." (ICRP publication 30, supplement to part 1.) *Annals of the ICRP*, 3:1-4, Pergamon Press, New York.

International Commission on Radiological Protection (ICRP). 1980. "Limits for intakes of radionuclides by workers." (ICRP publication 30, part 2.) *Annals of the ICRP*, 4:3-4, Pergamon Press, New York.

International Commission on Radiological Protection (ICRP). 1981a. "Limits for intakes of radionuclides by workers." (ICRP publication 30, part 3.) *Annals of the ICRP*, 6:2-3, Pergamon Press, New York.

International Commission on Radiological Protection (ICRP). 1981b. "Limits for intakes of radionuclides by workers." (ICRP publication 30, supplement to part 2.) *Annals of the ICRP*, 5:1-6, Pergamon Press, New York.

International Commission on Radiological Protection (ICRP). 1982a. "Limits for intakes of radionuclides by workers." (ICRP publication 30, supplement A to part 3.) *Annals of the ICRP*, 7:1-3, Pergamon Press, New York.

International Commission on Radiological Protection (ICRP). 1982b. "Limits for intakes of radionuclides by workers." (ICRP publication 30, supplement B to part 3, including addendum to the supplements of parts 1 and 2.) *Annals of the ICRP*, 8:1-3, Pergamon Press, New York.

International Commission on Radiological Protection (ICRP). 1988a. "Individual monitoring for intakes of radionuclides by workers: Design and interpretation." (ICRP publication 54.) *Annals of the ICRP*, 19:1-3, Pergamon Press, New York.

International Commission on Radiological Protection (ICRP). 1988b. "Limits for intakes of radionuclides by workers: An addendum." (ICRP publication 30, part 4.) *Annals of the ICRP*, 19:4, Pergamon Press, New York.

International Commission on Radiological Protection (ICRP). 1990. "1990 recommendations of the International Commission on Radiological Protection." (ICRP publication 60.) *Annals of the ICRP*, 21:1-3, Pergamon Press, New York.

International Commission on Radiological Protection (ICRP). 1994a. "Human respiratory tract model for radiological protection." (ICRP publication 66.) *Annals of the ICRP*, 24:1-3, Pergamon Press, New York.

International Commission on Radiological Protection (ICRP). 1994b. "Dose coefficients for intakes of radionuclides by workers." (ICRP publication 68.) *Annals of the ICRP*, 24:4, Pergamon Press, New York.

International Commission on Radiological Protection (ICRP). 1996. "Age-dependent doses to members of the public from intakes of radionuclides: Part 5 compilation of ingestion and inhalation dose coefficients." (ICRP publication 72.) *Annals of the ICRP*, 26:1, Pergamon Press, New York.

International Commission on Radiological Protection (ICRP). 1997. "Individual monitoring for internal exposure of workers." (ICRP publication 78.) *Annals of the ICRP*, 27:3-4, Pergamon Press, New York.

Jackson, S. (1966). "Creatinine in urine as an index of urinary excretion rate." *Health Phys.* 12:843.

Karpas, Z., A. Lorber, E. Elish, P. Marcus, Y. Roiz, R. Marko, R. Kol, D. Brikner, and L. Halicz. 1998. "Uranium in urine-normalization to creatinine." *Health Phys.* 74(1):86-90.

Kelso, S. M., and J. C. Wraight. 1996. "The measurement of aerosol size distributions (AMAD) in buildings on BNFL's Sellafield Site." *Rad. Prot. Dosim.* 63(2):1127-131.

Kim, E. M. 1995. *Determination of Time Interval of Urine Sample by Measuring Creatinine in Urine.* ESH-HPT-95007, Westinghouse Savannah River Company, Savannah River Site, Aiken, South Carolina.

Lessard, E. T., X. Yihua, K. W. Skrable, G. E. Chabot, C. S. French, T. R. La Bone, J. R. Johnson, D. R. Fisher, R. Belanger, and J. L. Lipsztein. 1987. *Interpretation of Bioassay Measurements*. NUREG/CR-4884, U.S. Nuclear Regulatory Commission, Washington, D.C.

Medley, D. W., R. L. Kathren, and A. G. Miller. 1994. "Diurnal urinary volume and uranium output in uranium workers and unexposed controls." *Health Phys.* 67(2):122-130.

National Bureau of Standards (NBS). 1953. *Maximum Permissible Amounts of Radioisotopes in the Human Body and Maximum Permissible Concentrations in Air and Water*. Handbook 52, U.S. Department of Commerce, Washington, D.C.

National Bureau of Standards (NBS). 1959. *Maximum Permissible Body Burdens and Maximum Permissible Concentrations of Radionuclides in Air and Water for Occupational Exposure*. Handbook 69, U.S. Department of Commerce, Washington, D.C.

National Council on Radiation Protection and Measurements (NCRP). 1987. *Use of Bioassay Procedures for Assessment of Internal Radionuclide Deposition*. NCRP Report No. 87, Bethesda, Maryland.

National Institute for Occupational Safety and Health (NIOSH). 1974. *Criteria for a Recommended Standard - Occupational Exposure to Benzene.* HEW publication (NIOSH) 74-137, U.S. Government Printing Office, Washington, D.C.

National Research Council. 1988. *Health Risks of Radon and Other Internally Deposited Alpha-Emitters*. BEIR IV, National Academy Press, Washington, D.C.

Nenot, J. E., and J. W. Stather. 1979. *The Toxicity of Plutonium, Americium, and Curium*. Commission of the European Communities, Pergamon Press, Oxford, England.

Palmer, H. E., R. W. Perkins, and B. O. Stuart. 1964. "The distribution and deposition of radon daughters attached to dust particles in the respiratory system of humans exposed to uranium mine atmospheres." *Health Phys.* 10:1129-1135.

Strenge, D. L., R. A. Kennedy, M. J. Sula, and J. R. Johnson. 1992. *Code for Internal Dosimetry (CINDY Version 1.2), Part 1: Conceptual Representation*. PNL-7493 Pt. 1, Rev. 1, Pacific Northwest National Laboratory, Richland, Washington.

U.S. Department of Energy (DOE). 1994. *Radiological Control Manual*. DOE/EH-0256T, Rev. 1, Washington, D.C.

U.S. Department of Energy (DOE). 1999. *Internal Dosimetry Program Guide for Use with Title 10, Code of Federal Regulations, Part 835, Occupational Rad. Prot.* DOE G 441.1-3, Washington, D.C.

U.S. Environmental Protection Agency (EPA). 1988. *Limiting Values of Radionuclide Intake and Air Concentration and Dose Conversion Factors for Inhalation, Submersion, and Ingestion.* Federal Guidance Report No. 11, EPA-520/1-88-020, Washington, D.C.