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HPS N13.22-1995

Foreword (This foreword is not a part of American National Standard HPS N13.22-1995.)

The purpose of this document is to provide minimum program standards for monitoring the internal exposure of workers to the various forms of uranium encountered in the workplace, using bioassay methods. "Bioassay," as used in this standard, means the measurement and analysis of quantities of radioactive material in human body compartments through either: (1) the direct detection of gamma rays emitted from the radioactive material within body compartments, using gamma-ray detectors external to the body (in vivo bioassay); or (2) the determination of amounts of radioactive material in samples from the body (e.g., excreta, blood, hair) and the evaluation of internal deposits of radioactive materials using appropriate mathematical models relating excretion rates or sample contents to amounts within body compartments (in vitro bioassay). This standard does not address programs for measuring uranium in air (air monitoring), or programs for monitoring contamination on surfaces or in other materials that might be taken into the body.

The Working Group has intentionally included some definitions and specifications that differ from those in relevant international standards; common American practices and regulatory requirements have dictated these deviations, which are considered minor and on the side of increased protection. The standard provides: criteria and conditions under which workers are to be included in a bioassay program; the frequencies with which bioassay sampling are to be carried out for monitored workers; quantitative action levels and actions that will help to ensure that workers will be protected against unacceptable levels of internal exposure, and that appreciable internal exposures will be appropriately monitored and recorded; and other aspects of bioassay programs needed to meet acceptable standards of internal exposure monitoring for uranium. Tables that are part of the standard are referenced in the text of the standard with Arabic numerals; these tables are placed immediately after the text of the standard.

Five appendices, with tables, figures, and references, are included with this standard but are not considered part of the standard. The appendices are included to provide the rationale for the quantitative criteria in the standard, and to provide a collection of information that could meet only the specifications of this standard. Also included is information on analytical methods and procedures for interpreting bioassay data in terms of internal dose. Tables, figures, and references that are cited in the appendices, and which are part of the appendices, are numbered with Arabic numerals preceded by capital letters associated with the respective appendices. Tables and figures associated with each appendix are placed immediately after the respective appendix. All references in the appendices are placed in a single list at the end of the document. Table numbers referenced in the appendices that are not preceded by letters are tables in the standard.

Existing NRC guidance in this area (Regulatory Guide 8.22)(RG 8.22) applying to uranium mills has been provided as a reference in the appendices. Some of the provisions of RG 8.22 might apply to other facilities that process materials similar to yellowcake and ore handled in mills. However, the Working Group decided that it was not necessary to incorporate provisions of RG 8.22 into this standard. RG 8.22 was developed over many years as a consensus of knowledgeable persons in the uranium mill industry, government, and the health physics profession. Also, provisions of RG 8.22 have already been imposed on the uranium milling industry. Much information incorporated into RG 8.22 came from early drafts of this standard. Recent studies tend to support the action levels and actions in RG 8.22 for uranium mills. Therefore, provisions of RG 8.22 have been affirmed by this working group and are considered to be appropriate for uranium mill bioassay programs.

In 1987, the Canadian government published guidance for bioassay programs involving various nuclidic and chemical forms of uranium. This guidance included detailed information for estimating internal exposures from intakes. In 1990, the International Commission on Radiological Protection (ICRP) published a complete revision of its reports 10 and 10A, as ICRP Report 54, with recommendations on the bioassay of many nuclides under conditions of routine exposure and accidental, single intakes. Recent data consistent with the ICRP 30 models were used to estimate derived investigation levels (DILs), including DILs for the various important uranium nuclides in D, W, and Y forms, according to ICRP 30 lung

dissolution rate classifications. However, this guidance can not be used for in vivo monitoring of highly enriched uranium. Other available guidance is in some cases more complex than necessary for use in establishing programmatic aspects of industrial bioassay in the United States. In some cases, guidance on the most recent human data on chemical toxicity has not been utilized to establish DILs for uranium in D and W transportability classes, and there is no consideration of special Class Y compounds.

The standard has been prepared with appropriate attention to the content and rationale of other standards and guidance documents in order to avoid unnecessary confusion among users. The working group has independently examined the radiobiological and toxicological information available on uranium compounds, and has attempted to provide DILs or action levels consistent with those in the already-developed documents, where the precision of data did not warrant substantial changes.

The Health Physics Society Standards Committee Working Group responsible for this standard had the following members*:

Allen Brodsky, Chairperson
(Georgetown University)

Michael A. Austin*
(U.S. Department of Energy)

James S. Bogard
(Oak Ridge National Laboratory)

Bruce Manninen
(Martin Marietta Energy Systems)

Stephen A. McGuire
(U.S. Nuclear Regulatory Commission)

E.R. Wagner
(Martin Marietta Energy Systems)

L. Max Scott
(Louisiana State University)

Casper Sun
(Brookhaven National Laboratory)

C.M. "Hap" West
(Oak Ridge Institute of Science and Education)

The members acknowledge significant contributions by the following consultants:

Roscoe M. Hall, Jr., was an important contributor to this standard before his death in 1991.

Robert E. Alexander – The Alexander Corporation
S. Robert Bernard – Oak Ridge, Tennessee
Richard K. Burklin – Siemens Power Corporation
J.J. Davis – Advanced Sciences, Inc.
Darrell R. Fisher – Battelle - Pacific Northwest Laboratory
Gary H. Kramer – Bureau of Radiation and Medical Devices, Canada
Milton E. McLain, Jr. – Texas A & M University
Michelle M. Reichert – Martin Marietta Energy Systems
Bob Robinson – General Electric Company
Charles T. Schmidt – University of California
Kenneth W. Skrable – Lowell University
J. Newell Stannard – San Diego, California
Paul S. Stansbury – Battelle - Pacific Northwest Laboratory
A.N. Tschaeché – Idaho Falls, Idaho
Robert A. Wessman – Thermo Analytical, Inc.

* Michael A. Austin died after this standard was forwarded to HPS N13 for consensus balloting.

This standard was developed under the direction of the Health Physics Society Standards Committee and approved on 28 November 1994. At that time, the Standards Committee had the following membership:

Chairperson: Harley Piltingsrud

Members: John Buchanan
Nancy Daugherty
Jack Fix
Nolan Hertel
David Kocher
Perry Moskowitz
Richard Toohey

Internal Dosimetry Section Chairperson: Gary Kramer

This standard was consensus balloted and approved by the ANSI-Accredited HPS N13 Committee on 15 June 1995. At the time of balloting, the HPS N13 Committee had the following membership:

Chairperson	Jerre Forbes
American Chemical Society	Al Zirkes
American Industrial Hygiene Association	George Wilkening
	John Masaitis (alt.)
American Iron and Steel Institute	Anthony LaMastra
	Peter Hernandez (alt.)
American Mining Congress	Scott Munson
American Nuclear Insurers	Jerre Forbes
American Nuclear Society	Joyce Davis
American College of Occupational and Environmental Medicine	Bryce Breitenstein
Conference of Radiation Control Program Directors	Steve Collins
Edison Electric Institute	Robert Sorber
	Matthew Mingoia (alt.)
Health Physics Society	Kenneth Swinth
Institute of Electrical and Electronic Engineers	Lou Costrell
Institute of Nuclear Materials Management	Kenneth Okolowitz
National Council on Radiation Protection and Measurements	James Spahn, Jr.
Nuclear Management and Resources Council	John Schmitt
U.S. Department of Commerce	Thomas Hobbs
	Lester Slaback, Jr. (alt.)
U.S. Department of Energy	Robert Loesch
	Joel Rabovsky (alt.)
U.S. Department of Defense	Isaac Kunz
U.S. Environmental Protection Agency	Allan Richardson
U.S. Nuclear Regulatory Commission	Donald Cool
	Jay Cunningham (alt.)
U.S. Public Health Service	Edward Tupin
U.S. Navy	Karl Mendenhall
Oil, Chemical and Atomic Workers International Union	Dean Alexander
	Charles Barrett (alt.)
Individual	John Auxier
Individual	George Campbell
Individual	Hugh Henry
Individual	Ronald Kathren
Individual	Edward Reitler, Jr.
Individual	L. Max Scott
Individual	Al Tschaeche
Individual	McDonald Wrenn

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Bioassay Programs For Uranium

1. Introduction

1.1 General Considerations

Each facility that handles unencapsulated uranium in significant quantities requires a radiation protection program to help minimize or prevent, to the extent reasonably achievable, internal exposure to uranium. Such exposure can occur via inhalation, absorption through the skin or through breaks in the skin, or by ingestion. This program should include measurements to estimate quantities of uranium taken into the human body and the resulting exposures to different body organs, or to establish that significant intakes have not occurred. These measurements, which include urine sampling, fecal sampling, and *in-vivo* counting, are referred to as "bioassay" in radiation protection practice. Bioassay measurements are analyzed and interpreted to evaluate internal doses to personnel. Also included as part of a bioassay program are "action levels," or the values of uranium bioassay measurements at which certain steps are taken to investigate sources of exposure and/or reduce further exposure.

Each facility that processes unencapsulated uranium provides engineered safety features, such as ventilation systems and process enclosures. These safety features are designed to protect both workers and members of the public from exposure to hazardous levels of uranium. Safety features have been considered in establishing standards for the bioassay programs for each type of facility.

The body of this standard provides a consensus of acceptable criteria for establishing bioassay programs for different forms of uranium. Additional information useful in interpreting and acting upon bioassay data are included in the

appendices. The appendices are not part of the standard.

The following statements of Purpose and Scope describe the specific contents of this standard.

1.2 Purpose

This standard provides criteria for establishing and managing a bioassay program to monitor and evaluate intakes from uranium, distributions of uranium within the body following intake, and the resulting radiation doses or possible chemical effects. Action levels are also provided in terms of measured bioassay quantities, in order to help ensure, for various uranium compounds and isotopic enrichments, that exposure to workers from internally deposited uranium will be maintained below acceptable limits.

1.3 Scope

This standard provides bioassay program requirements for long-lived uranium isotopes (half-life greater than ^{234}U), including criteria for determining:

- a) the conditions requiring bioassay;
- b) the selection of individuals to be included in a bioassay program;
- c) the sampling and measurement frequencies; and
- d) the action levels and corresponding actions that are appropriate for different compounds, measurements, and sampling frequencies.

Procedures for quality assurance and records requirements are provided in other bioassay program standards, and in applicable federal, state, and local regulations.

The appendices to this standard are not a part of the standard. However, they provide the bases and rationale for the quantitative provisions of the standard. They are also provided to assist in establishing bioassay programs by presenting technical information, procedures, and literature references that are helpful in developing the sample collection and analytical procedures necessary to bioassay programs. The appendices also include guidance for the estimation of internal radiation doses from bioassay data, and methods of estimating internal chemical exposures for comparison with toxic limits.

This standard does not specify preferred chemical, radiometric, or *in-vivo* counting procedures for measuring uranium. This standard also does not specify quantitative performance criteria for the accuracy and precision of analytical or measurement procedures; another standard is referenced for these criteria.

2. Definitions

There are many terms and definitions associated with uranium bioassay programs in American industry that have special meaning to this industry. The following definitions of terms are part of this standard. They are for use with this standard, but may in some cases differ from definitions in use, or under development, in other related standards. Since many pertinent national and international standards are continually under review and revision, some of these definitions may be subject to change in future revisions of this standard.

Absorbed Dose (D). Energy absorbed per unit mass of irradiated material. The SI unit for the quantity absorbed dose is the gray (Gy):

$$1 \text{ Gy} = 1 \text{ J kg}^{-1} = 100 \text{ rad} = 10^4 \text{ erg g}^{-1}$$

Action Level. A quantity of radioactivity in excreta or in the body at which certain defined actions are taken to investigate exposure conditions or restrict further exposures.

Activity Median Aerodynamic Diameter (AMAD). The diameter of a unit density sphere with the same terminal settling velocity in air as that of the aerosol particle whose activity is the median for the entire aerosol.

Acute Exposure. Exposure to uranium intake of a short duration.

ALARA. As Low As Reasonably Achievable, economic and social factors being taken into consideration.

Annual Limit on Intake (ALI). A recommended limit on the annual intake (in Bq) for an individual radionuclide; it is a secondary limit based on calculations that limit the internal committed effective dose equivalent to the standard recommended by the ICRP (ICRP 30).

Background Urinary Uranium. An excretion rate in urine equivalent at equilibrium to the average food and water intake of natural uranium. This value varies for different locations and food sources.

Becquerel (Bq). The SI unit of activity; 1 Bq = 1 disintegration s^{-1} .

Bioassay. The determination of the kind, quantity, location, and/or retention of radionuclides in the body by direct (*in vivo*) measurement or by indirect (*in vitro*) analysis of material excreted or removed from the body.

Biokinetic Model. A set of mathematical relationships formulated to relate the intake to the uptake, distribution, and retention of a radionuclide in various organs and tissues of the body. Some models include subsequent excretion from the body by various pathways. **Note:** A biokinetic model is used in ICRP Publication 30 in conjunction with an intake model to estimate dose per unit intake of particulates containing radionuclides. Sometimes the term "metabolic model" is used to connote "biokinetic model."

Breathing Zone Air Sampler (BZA). This term is used for an air sampling device that collects airborne nuclides in a work place location near the point of a worker's inhalation. This device is

designed to quantify a worker's inhalation exposure.

Committed Effective Dose Equivalent (CEDE). The summed products of committed dose equivalents to individual tissues times the respective tissue weighing factors. (See ICRP 30 and ICRP 42.)

Confirmatory Bioassay. This term has often been used to connote bioassay measurements to confirm the validity of previous measurements. The term is also used in this standard as used in ICRP 54, to confirm that intakes or exposures are below certain limits or action levels (see ICRP 54).

Control Level. A quantity of radioactivity in the body or excreta at which exposure is limited by such actions as correcting the cause of elevated air levels, controlling access to the area of most significant air contamination, or wearing of respiratory protection.

Chronic Exposure. Exposure to uranium intake of long duration, delivered by fractionation or protraction.

Class D, W, or Y Material. ICRP Publication 30 classifies inhaled radioactive materials as D, W, or Y (days, weeks, or years) depending on their retention time in the pulmonary region. Class D materials have a pulmonary half-time of less than 10 days; W materials, a half-time from 10 to 100 days; and Y, greater than 100 days. The times actually used for these classes of materials in ICRP 30 in the calculation of ALI's are 0.5 days, 50 days and 500 days for Class D, W, and Y material, respectively. Note: A Special Class Y uranium has been defined as having an effective pulmonary half-life of 100 days and the systemic distribution, retention, and excretion parameters of Class W uranium (see Section 6.5 and Table 7).

Decision Level. A net count of activity at which it is decided that a positive detection of activity above an appropriate blank has been obtained, with a specified probability (see Appendix D).

Depleted Uranium (DU). Depleted uranium is a byproduct of the enrichment process. Depleted uranium has a reduced abundance of ^{235}U relative to the abundance of 0.7 % in natural uranium. The typical ^{235}U contents for depleted uranium and natural uranium are 0.2 % and 0.7 %, respectively. Depleted uranium also has a reduced abundance of ^{234}U .

Derived Air Concentration (DAC). The concentration in air (Bq m^{-3}) that if inhaled by reference man for 2000 h (50 wk, 40 h wk⁻¹, 0.02 m³ min⁻¹; or 2400 m³ y⁻¹ inhalation rate) during employment with average effort, would result in the ALI by inhalation:

$$\text{DAC} = \text{ALI}/2.4 \times 10^3 \text{ Bq m}^{-3}$$

Derived Investigation Level (DIL). A measured quantity, derived by calculations from a basic dose level, at which additional monitoring, investigation, or exposure controls are to be carried out (as defined in ICRP 54).

Derived Level (DL). A quantity derived from appropriate dose-equivalent limits or annual limits of intake by a defined biokinetic model in such a way that compliance with the DL implies virtual certainty of compliance with relevant basic internal dose limits.

Dose Equivalent (DE) or (H). Defined by

$$H = D Q N,$$

where

- D = absorbed dose from a particular radiation,
- Q = quality factor for a particular radiation (more recently called the "radiation weighing factor" by the ICRP -- see ICRP 60),
- N = the product of all other modifying factors specified by the ICRP.

The ICRP has assigned a value of unity to N for the present. The value for Q is assumed to be 20 for alpha particles and 1 for photons and electrons. The SI unit of dose equivalent is the same as for absorbed dose, J kg⁻¹, in fundamental metric units; however, the unit is termed the Sievert (Sv), which has different

implications regarding radiation risks than the gray. (See ICRP 26 and ICRP 60.)

Enriched Uranium. Uranium with an increased abundance of ^{235}U . Enriched uranium varies from the natural ^{235}U abundance of 0.7% up to almost 100%. Enriched uranium also contains increased abundances of ^{234}U and might also contain ^{236}U and ^{233}U .

Evaluation Level. An activity level in the body or excreta at which the measurement results are examined for validity by a review of workplace monitoring results or additional bioassay monitoring. (The magnitudes of this level differ in ICRP 54 and have different nomenclature in that document.)

ICRP. An abbreviation for the name of the International Commission on Radiological Protection, an international organization formed by the International Congress of Radiology in 1928 to develop recommendations for protection against ionizing radiation. Reports of the ICRP that are pertinent to this standard have been referred to by report number; these reports are listed in the references presented in the last appendix to this standard.

Implementation Level (or "Program Implementation Level"). A level of potential exposure as estimated from air monitoring, or a quantity of material in process, above which a bioassay program is to be implemented.

Intake. (1) The ingestion or inhalation of a radioactive material. (2) The amount of radioactive material taken into the body by inhalation, ingestion, absorption through the skin, injection, or via a wound.

Investigation Level. An activity level in the body or excreta at which the measurement results are examined for validity and investigation is made to see whether exposures can be reduced ALARA, as used in ICRP 54.

Minimum Bioassay Program. A limited bioassay program designed to sample only a few most-exposed workers to confirm that safe routine operations are maintained, or to monitor

levels of exposure below $0.1 \text{ ALI } \text{y}^{-1}$. It may consist of only confirmatory bioassays.

Minimum Detectable Amount (MDA). The smallest amount (activity or mass) of an analyte in a sample that will be detected with a probability β of non-detection (Type II error) when the decision level for detection accepts a probability α of erroneously deciding that a positive (non-zero) quantity of analyte is present in an appropriate blank sample (Type I error). For this standard, α and β probabilities are both set at 0.05. Concepts and examples of MDA for uranium measurements are discussed further in Appendix D.

Monitoring. Measurement of radioactivity for reasons related to the estimation or control of exposure to radiation or radioactive materials. The term includes the interpretation of measurement results.

Natural Uranium. The mixture of uranium as it is found in nature, i.e., 99.3% ^{238}U , 0.7% ^{235}U , and 0.0058% ^{234}U by weight. The alpha activity comes mostly, and in about equal amounts, from ^{238}U and ^{234}U . The specific activity of this mixture is $2.6 \times 10^7 \text{ Bq kg}^{-1}$ ($0.7 \text{ pCi } \mu\text{g}^{-1}$).

Reference Man. A person with the anatomical and physiological characteristics defined in the report of the ICRP Task Group on Reference Man (ICRP Publication 23).

Retention Function. A mathematical expression for the fractional retention of a nuclide in an organ, tissue, body, or excretion compartment at any time. It can be a fraction of the intake, or a fraction of the uptake in body fluids or in the compartment of interest, i.e., either an intake retention function (IRF) or an uptake retention function (URF).

Routine Uranium Bioassay Program. A regular program of collecting and measuring bioassay samples in order to monitor and estimate exposures to workers who are exposed, or potentially exposed, to uranium during their routine work activities, and to help ensure that limits of intake are not exceeded. A routine bioassay program should also include

pre-placement or *pre-work* bioassay measurements to establish "baseline" levels not related to the new work environment, as well as *termination bioassay* measurements to establish exposure status at the termination of employment, or at the end of work at a specific work station or area.

Secular Equilibrium. If a radionuclide has a very much longer half-life than its decay products (so that there is no appreciable change in its radioactivity in the time interval required for the activity of all decay products to attain equilibrium with the activity of the parent radionuclide) then, after equilibrium is reached, equal numbers of atoms of all members of the series disintegrate per unit time.

Service Laboratory. The in-house or vendor laboratory that routinely performs *in-vitro* or *in-vivo* analyses on employees.

Special Bioassay. Measurements following specific known or suspected acute exposures to determine amounts of exposure more reliably, sometimes referred to as "diagnostic" bioassay in previous standards.

Testing Laboratory. The external laboratory that provides standards or spiked samples with known amounts of uranium for the external quality control program of the service laboratory.

Uptake. (1) The passage of material from the location of intake into the body fluids. (2) The quantity of radioactive material taken up by the systemic circulation. (3) The quantity of radioactive material taken up by a specific body organ.

Uranium Urinalysis (urine analysis, urine bioassay, urine assay). Analysis of urine samples for the presence of uranium.

Work Restriction Level. An activity level in the body or excreta at which action is taken to prevent further exposure by removing the employee from areas of potential exposure until exposure evaluation is completed and the exposure rate is such that the employee is approved to return to uranium work. See Section 6.

Note: Special terms used in uranium mill bioassay programs are presented in Regulatory Guide 8.22 of the U. S. Nuclear Regulatory Commission.

3. Establishing the Need for an Internal Dosimetry Program

3.1 General Considerations

A bioassay program capable of detecting exposures of employees should be planned and implemented whenever an employee is potentially exposed to intakes of uranium approaching levels that expert national or international bodies have recommended be measured and recorded in occupational exposure records. Routine bioassay programs shall be implemented whenever potential exposures might exceed any recommended implementation or action levels. It is not always possible, before an operation has begun, to predict expected levels of exposure. Therefore, in this implementation section, conservative criteria are given to implement programs based on the quantities of uranium handled or processed by workers. The criteria take into account recommended primary investigation levels or derived action levels. (Conservative criteria are considered to be those that would be more likely to result in implementing a program that was not needed than to not implement a program that was needed in the interest of safety.) The criteria based on amounts in process are derived from considerations discussed in Appendix A. Sections A.1 and A.2 relate amounts of uranium in process to practical upper limits of exposure, based on observations and experience from various operations using unsealed radioactive materials.

3.2 Program Implementation Criteria

3.2.1 Conditions Under Which Bioassays Should Be Performed for Uranium

3.2.1.1 Routine bioassay programs should be established for all employees who routinely work in areas, or are directly involved in tasks in which there is potential uranium exposure, as given by

the criteria in the following sub-sections. Baseline (or pre-placement) bioassays should be performed prior to initial assignments for such work.

3.2.1.2 Bioassay shall be performed as a minimum when workers are likely to receive an annual intake in excess of 10% of the applicable ALI. In addition, bioassays shall be performed when intakes of 1 mg or more of soluble uranium are likely to occur in any one work day, to ensure that adverse chemically toxic effects are unlikely.

3.2.1.3 Whenever worker intakes or air concentrations cannot be reliably predicted or measured (such as by air monitoring or evaluation of ventilation and containment parameters), routine bioassay programs shall be implemented whenever quantities of uranium handled (or in process within the breathing zone of any worker) exceed the respective quantities in Table 1. A confirmatory bioassay program (e.g., a minimum bioassay program consisting of an appropriate sampling of workers to supplement the monitoring of airborne uranium in the environment) should be implemented at levels exceeding 0.1 of those in Table 1. This confirmatory program should supplement, not replace, the appropriate air monitoring program.

3.2.1.4 If respiratory protection is used to maintain inhalation exposures below any action levels, bioassay shall be performed to verify the effectiveness of the respirators.

3.2.2 Conditions Under Which Bioassays May Be Discontinued

When conditions in Section 3.2.1 requiring bioassay no longer exist, then the bioassay program for the respective individual(s) may be discontinued. In these cases, and in cases where the employee's work in uranium areas is terminated, appropriate termination bioassays should be performed after the last exposure. A minimum workplace monitoring program should still be maintained to ensure that controls remain adequate.

4. Frequencies of Bioassay

4.1 General Considerations

4.1.1 Frequencies of bioassay should be based on the ability of analytical methods to detect the derived levels (DL) of uranium, in excreta samples or in regions of the body, that would lead to significant internal doses to workers. In this standard, internal doses are deemed of sufficient significance to require detection when they are expected (on the average) to reach the evaluation levels (EL) in this standard. Thus, the frequencies of bioassay are to some degree inter-related to the chosen ELs and the minimum detectable amounts (MDA) of the respective analytical procedures. (See Section 6.1, Appendix A.3 and guidance in Appendix D on estimating MDAs.)

4.1.2 Use of the concept of basing sampling frequencies on action levels calculated from air monitoring should be weighed against the "missed dose" concept (see Appendix A.3.10) and other dosimetry, administrative, or regulatory concerns in consideration of the fact that the best overall exposure control comes from a well-balanced program. Such a program includes air monitoring and contamination surveys as well as bioassays. It is within that context that minimum sampling frequencies have been established for determination of the action points listed in Section 6 and Tables 2-7.

4.2 Frequencies

4.2.1 Minimum Frequencies for Routine Bioassay Programs

As stated in Section 4.1, the frequency of bioassay for routine bioassay programs is interrelated with action levels. Consequently, minimum frequencies are specified in conjunction with action levels. Table 8 summarizes the minimum frequencies specified in Section 6.

4.2.2 Other Frequency Situations

There can be situations where it would be necessary to sample more frequently than is indicated in Table 8. If that is the case, the action levels prescribed in Section 6 and its associated tables should be re-evaluated. However, the derived action levels can be used for more frequent monitoring than specified above under many circumstances. (See also Sections 6.4, 6.6, and Appendix A.3.5.)

4.2.2.1 For chronic exposures to soluble uranyl compounds (those leaving the body in hours or days) approaching the occupational exposure limits, more frequent bioassays than suggested in 4.2.1. should be taken. It may be appropriate to sample after each work break, or at the beginning and end of the work week, to best estimate individual exposures and help ensure that overexposures do not occur under such circumstances.

4.2.2.2 Lung burdens resulting from exposure to pure class Y uranium can be detected (and dose can be estimated) only following extremely high acute or elevated long-term chronic exposures. Tables 4 and 6 show that levels at which exposure control actions should be taken for exposure to class Y uranium are less than the minimum detectable levels attainable for single sample measurements. Consequently, monitoring may be done either by fecal analysis, or urinalysis methods with lower MDAs, if exposure to pure class Y material occurs. An alternative approach utilizing urinalysis to monitor these exposures might be more frequent monitoring, resulting in a lower MDA for the average of a number of measurements, rather than a single measurement. This method for improving MDAs is presented in Appendix D.

4.2.2.3 Exposure to both pure Class Y and to more transportable classes of uranium during the same monitoring period presents additional difficulties, since no radiobioassay method discriminates among the inhalation classes of material. In these cases, auxiliary methods of determining transportability of the uranium mixtures in tissue fluids (e.g., methods obtained from the literature) may be required for

calculating appropriate bioassay frequencies and action levels. Some of the information provided in the appendices to this standard has been retained for its possible value in such calculations.

5. Selection of Individuals for Bioassay

5.1 All individuals who are exposed to levels of uranium above those in Section 3 that require routine bioassay programs shall be sampled at frequencies indicated in Section 4.

5.2 Individuals exposed to levels of uranium for which confirmatory bioassay is recommended in Section 3 may be sampled less frequently than specified in Section 4. A suitable sample of these individuals may be monitored at the frequencies specified in Section 4.

5.3 A policy regarding participation of employees in the bioassay programs shall be established at any facility having such programs. Those establishing the programs should consider making mandatory participation a condition of employment.

6. Action Levels and Follow-Up Actions

6.1 General Derivation of Action Levels

6.1.1 Action levels were developed based on an adaptation of the approach suggested in ICRP Publication 54. (See Section A.3 of Appendix A.) Action points are set at $1/10$ ALI(T/365)m(T/2), $3/10$ ALI(T/365)m(T/2), and ALI(T/365)m(T/2) as Evaluation, Control, and Work Restriction levels, respectively, for the frequency chosen as the minimum, where:

ALI is the Annual Limit on Intake prescribed by ICRP 30;

T is the time of the sampling interval (i.e., time between samples) in days; and

m(T/2) is the fraction of a single intake taken in at $t=T/2$ that will remain in the urine (or in the lung) at $t=T$ days after the

previous sample. A special adaptation of this approach was also used with intakes based on chemical toxicity, which was not addressed in ICRP 30.

6.1.2 The action levels derived as presented in Section 6.1.1 are given: In Table 2 for urinalysis applicable to different forms of uranium, in Table 3 for fecal analysis applicable to the intake of Class Y uranium compounds, and in Table 4 applicable to *in-vivo* counting of classes W and Y for ^{235}U and ^{238}U . When bioassay monitoring results exceed any of the action levels in Tables 2 through 4, the respective actions indicated in the tables shall be taken, as specified in the following Sections 6.2 and 6.3.

6.2 Actions Applicable to Routine Urinalysis, Fecal, and In-Vivo Bioassays

The actions below are applicable to the action levels in Tables 2 through 4. (Although, as indicated in Section 6.1.1, the respective action levels were derived using an adaptation of methods used in ICRP 54, the names of the action levels have been changed. In this standard, the actions to be taken have been made consistent with American practice requirements.)

6.2.1 Evaluation Level (EL) Actions.

The radiation protection organization (RPO) shall evaluate conditions of exposure to determine if exposure is authentic. Such an evaluation may include a review of workplace monitoring data and the collection of additional bioassay data.

6.2.2 Control Level (CL) Actions. In addition to the actions above, the RPO shall evaluate possible exposure control actions. These actions should include the estimating of intakes and also may include: improving engineering controls; establishing additional administrative controls; and the use of personal protective equipment. The exposure history and the frequency of routine measurements for the worker also should be reviewed. The employee and the employee's supervisor should be notified of estimated intakes or exposures. If samples consistently exceed this Control Level, or

approach the Work Restriction Level, the RPO may consider some of the actions listed in section 6.2.3 below.

6.2.3 Work Restriction Level (WRL) Actions.

Actions listed under Section 6.2.2 shall be taken along with the following actions to *prevent* additional intake and to further evaluate exposure. Written notice shall be sent to the supervisor and plant physician recommending that the employee be removed from any areas of potential exposure until exposure evaluation is completed and the employee is approved to return to the work area. A series of special bioassay measurements shall be made on the involved employee after he/she is removed from the area of exposure potential to determine individual retention and excretion parameters. The results from the bioassay determinations shall be used in conjunction with other pertinent information, to obtain estimates of intake and dose.

6.3 Conditions for Adjustment of Action Levels

These action points shall be reviewed for adequacy of overall exposure control, when employees are potentially exposed to other radiation sources or toxic agents. Since uranium has both chemical and radiobiological toxicity characteristics, urinalysis results should be interpreted both in terms of mass and radioactivity to ensure that the most appropriate set of action levels is used. (See Appendix A.3.) In addition, interpretation of urine and fecal analyses for class Y uranium shall take into consideration the normal excretion levels of uranium, as excreted by persons not exposed to uranium in the work environment.

6.4 Actions for Special Bioassays

Routine bioassay action levels in Tables 2 through 4 should be used only when average air concentrations are within occupational limits (DACs). Air sampling results should be reviewed on a timely and routine basis to help assure that significant exposures are evaluated by bioassay measurements. In order to help evaluate higher exposures from an accident or

incident that could have caused an intake of Class D or W uranium, special bioassay urinalysis monitoring shall begin as soon as feasible after any acute exposure. (Immediate fecal monitoring, with complete collection over several days, will also be helpful, when feasible, in obtaining better estimates of intake, even for Class D and W uranium.) The employee should be evaluated further if any of the action levels of Table 5 are exceeded. These action levels are based on a 2-mg d⁻¹ limit of intake for Class D and an 8-mg d⁻¹ limit for Class W.

6.5 Action Levels for Special Class Y Material

Table 6 shows that the control levels for urinalysis in monitoring Class Y materials are below routinely attainable minimum detectable activity (MDA) concentrations for many laboratories. One approach to this problem would be to develop and use methods that have a lower MDA than that routinely attainable.

Another approach that may be used by many, if not all, facilities would be to classify UO₂ and U₃O₈ as Special Class Y material with approximately a 100-d half-life in lung, with excretion parameters the same as Class W material. [This approach would fit the experience at many facilities (e.g., see references ICRP 54, Schieferdecker et al., West et al., and Forrest and Barber).] This approach would give the urinalysis action levels in Table 7. The action levels for a Special Class Y material may be used only if supported at a given facility by excretion data substantiating a different clearance rate than used in the standard model (ICRP 54). (See Appendix A.3.10.)

Table 8 summarizes the bioassay frequencies upon which action level calculations have been based.

6.6 Special Fecal Bioassay Action Levels for Class Y Material

In addition to their usefulness in a routine bioassay program, fecal analysis results can be extremely useful on samples taken immediately after an incident or accident for the evaluation of

intake. Table 9 presents the fraction of initial intake excreted in 24-h fecal samples and accumulated fecal excretion in the first 7 d post-intake. Table 9 may also be used to establish appropriate action levels for special bioassay. In these analyses, fecal clearance of uranium taken into the body as a result of occupational exposure must be discriminated from naturally occurring uranium from the individual's diet, using appropriate pre-occupational fecal blank analyses where available.

6.7 In-Vivo Action Levels for Class W Material

The control levels (CL), as shown in Table 4, are lower than MDAs routinely achievable in bioassay programs by current technologies. The CL in Table 4 represent the total alpha activities of the uranium isotope. In view of the limitations of *in-vivo* monitors, any confirmed lung count at or above detectable levels should trigger the appropriate CL action, as indicated in Table 4 and Section 6.2. The total calculated alpha activity relative to measured quantities of ²³⁵U or ²³⁸U will vary according to the isotopic mixture. The examples in Table 4 are for the specified compositions. Action levels should be developed by a facility for the particular isotopic composition of material handled or processed. (While not addressed in this standard, air sampling should also be used for monitoring and establishing workplace controls for employee exposure.)

6.8 Other Action Levels

Other action levels and actions may be derived on a specific case basis if the methods, data, literature, and standards used in derivation are documented. An example of such an approach would be the action levels developed by the U.S. Nuclear Regulatory Commission for uranium mills. (See references in Appendix E.) Another example is the approach for action levels for exposures to UF₆ and UO₂F₂ aerosols based on analyses of the Sequoyah accident (see Appendix B.3). Action levels lower than those suggested in this standard may be used for administrative purposes, such as to make an ALARA program more effective.

7. Quality Assurance and Control

7.1 Each facility should maintain a continuing quality assurance and control program on all facets of its bioassay program, including such items as sample collection, qualifications of laboratory personnel, laboratory inter-comparisons, routine control programs, computational checks, and use of appropriate blanks and standards.

7.2 Bioassay should be accomplished in an accredited laboratory or *in-vivo* facility.

7.3 Quality assurance and control programs should meet specifications prescribed by appropriate standards, especially in regard to: use of national standard reference radioactive material for calibrations and yield determinations; use of standard methods of determining minimum detectable activity and accuracy; written procedures for laboratory management, analyzing samples, and calculating results; and participation in interlaboratory analytical comparisons and applicable laboratory accreditation programs at appropriate frequencies. Until superseded by a final standard, the draft ANSI N13.30 standard listed in Appendix E (References) may be considered applicable to the provisions of this section. The final standard should be applied when available.

7.4 In the event that an external service laboratory performs the actual analyses and measurements of uranium upon which a bioassay program is based, the standard quality assurance and quality control requirements should be imposed on this external service laboratory by an appropriate contractual arrangement.

8. Information to be Included in the Records

8.1 The data used in evaluations, and the results of *all* bioassay measurements, should be included in internal dosimetry records, as specified in other applicable standards.

8.2 All information applicable to internal dosimetry in standards on recordkeeping in radiation protection, as published by the American National Standards Institute, shall be entered into the record and maintained as prescribed in the applicable standards.

9. Exposure and Dose Interpretation

9.1 All sample results for the period covered should be used when an estimate of internal dose is made.

9.2 When bioassay sample results are well below any of the Work Restriction Levels in Tables 2 through 7, intake and dose commitment estimates may be made using simplified methods such as those in Appendix B, assuming parameters for ICRP Reference Man.

9.3 When bioassay sample results are at or above 70% of the Work Restriction Levels, estimates of intake and dose commitment should be made using metabolic models such as those discussed in Appendix B, or improved models in more recent literature, fitted to sequential bioassay sample data on the individual employee in order to obtain more accurate estimates of dose received. The calculations and results shall be included in the employee's exposure records.

9.4 When individual bioassay sample measurements are below the decision level, then individual sample results shall be recorded together with their standard errors and allowed to accumulate in the individual employee's records. Individual recorded results shall *not* be rounded to zero or any "less than" statement; but neither shall they be used individually to make intake or dose commitment estimates. (See Appendix D.)

9.5 Before any bioassay result is used for an intake or dose commitment determination, the pre-employment (baseline or background) amount of uranium in an individual determination should be subtracted from the laboratory's reported individual sample result. If an appreciable amount of uranium is present in an

individual's lung or excreta from previous intakes of uranium, then estimates of the amounts present from these previous intakes should be made using appropriate intake and retention functions (see Appendix B.2 to B.4., and Tables

B.1 and B.2), and should be subtracted from sample results before the intake (and resulting dose commitment) attributed to the work period for which the dose is being estimated is calculated.

Table 1 – Implementation Levels: Mass or Activity Levels Above Which at Least Minimum Uranium Bioassay Programs Shall be Implemented^a

Types of Operation	Mass	Activity Amount ^c
Processes in open room on bench top, with possible escape from process vessels	0.5 kg ^b	1.2×10^7 Bq (320 μ Ci)
Process with possible escape of uranium that are carried out within a fume hood of adequate design, face velocity, and performance reliability	5 kg	1.2×10^8 Bq (3,200 μ Ci)
Processes carried out within glove boxes that are ordinarily closed, but with possible release from process vessels and occasional exposure to contaminated box and leakage	50 kg	1.2×10^9 Bq (32,000 μ Ci)

^a Values have been chosen to be conservative for any transportability class or admixture of isotopes of uranium. For a particular type of operation, the value of mass or activity that is more restrictive for the isotopic admixture of uranium to be handled should be used.

^b Obtained as shown in Appendix A.1.

^c Obtained from DAC values in Table A.2 for pure ²³⁵U, using relationship discussed in Appendix A.2.

Table 2 - Urinalysis Action Levels^a

I. Class D Uranium

A. Radiological Toxicity

Method: Urinalysis

Minimum frequency: Monthly

Actions ^a	Action Levels as Concentrations in Urine	
	(Bq L ⁻¹)	(pCi L ⁻¹)
Evaluation	1.5	40
Control ^b	4.4	120
Work Restriction ^c	15	400

^a Action levels and actions are defined in Section 6.1.^b Same as ICRP 54 DIL. Numbers in pCi L⁻¹ are rounded.^c If confirmed by three samples over the period of a month, which should be the maximum length of sampling interval.

B. Chemical Toxicity - Class D and W

Method: Urinalysis

Minimum Frequency: Monthly

Actions	Action Levels as Concentrations in Urine
	(µg L ⁻¹)
Evaluation	10
Control	30
Work Restriction ^d	90

^d If confirmed by at least three samples over a quarter.

II. W Class Uranium - Radiological Toxicity

Method: Urinalysis

Minimum Frequency: Quarterly

Actions	Action Levels as Concentrations in Urine	
	(Bq L ⁻¹)	(pCi L ⁻¹)
Evaluation	0.26	7
Control	0.74	20
Work Restriction ^d	2.6	70

^d If confirmed by at least three samples over a quarter.

Note: See Table 6 for action levels for Class Y uranium, compared with typically detectable concentrations in urine.

Table 3 - Fecal Analysis Action Levels

Y Class Uranium
 Method: Fecal
 Minimum Frequency: Quarterly

Action Levels ^a	Concentrations in Feces	
	(Bq d ⁻¹)	(pCi d ⁻¹)
Evaluation	0.0074	0.2
Control	0.019	0.5
Work Restriction	0.074	2

^a Action levels are further defined in Section 6.1.

Table 4 – Comparison of the Minimum Detectable Activity of *in-vivo* Measurement for Normal and Fully Enriched Uranium^a with Action Levels

Description	Action Level or MDA	Total U Alpha Activity ^b (Bq)	Activity ^b - Isotope
Class Y, Fully Enriched	Control Level	75	2.35 Bq – ²³⁵ U
Class W, Normal	Control Level	135	65.7 Bq – ²³⁸ U
Fully Enriched	MDA, Typical	142	4.48 Bq – ²³⁵ U
Normal Uranium	MDA, Typical	144	69.7 Bq – ²³⁸ U
Class Y, Fully Enriched	Work Restriction Level	250	7.83 Bq – ²³⁵ U
Class W, Normal	Work Restriction Level	450	218.7 Bq – ²³⁸ U

^a For the purpose of this conversion, fully-enriched uranium was considered to be 93% ²³⁵U; 6% ²³⁸U; and 1% ²³⁴U by mass.

^b The Total Uranium Alpha Activities quoted above in the third column are based on fully-enriched and normal uraniums. To determine the total alpha activity for any particular situation, the specific activity for the mix of uranium isotopes involved in the exposure needs to be calculated. To do this, first sum the products of the weight fraction of each isotope involved multiplied by its specific activity; then multiply the MDA in activity units of ²³⁵U or ²³⁸U (as judged from its decay products) by the ratio of the total alpha activity to the alpha activity of ²³⁵U or ²³⁸U.

An example: The total alpha activity for an action level of ²³⁵U for a uranium which is 60% ²³⁵U by weight and has a specific activity of 10⁹Bq kg⁻¹ could be calculated as follows:

$$\text{MDA} \times \text{ActivityRatio} = \text{Total associated alpha activity}$$

$$4.48 \text{ Bq of } ^{235}\text{U} \times \frac{10^9 \text{ Bq/kg}}{1 \text{ kg.} \times 0.6 \times 8 \times 10^7 \text{ Bq/kg U-235}} = 93.3 \text{ Bq}$$

Table 5 – Levels for Special Bioassay and Chemical Toxicity Evaluation^a

Time in Days After Acute Intake	Investigation Levels as Concentrations in Urine ($\mu\text{g L}^{-1}$)	
	Class D ^b	Class W ^b
1	270	245
2	94	62
3	41	27
4	24	18
5	19	15
6	16	14
7	14	13

^a Levels are based on chemical toxicity and are to be used for Class D or W uranium compounds, after acute (single intake) exposures of 2 mg and 8 mg, respectively. In this table, numbers have *not* been rounded because they are calculated numbers based on relative amounts excreted according to the ICRP model as shown in ICRP 54.

^b The levels listed for these values are so similar, it may be advantageous to use Class W limits for both types of material rather than to classify the exposure material relative to solubility class.

Table 6 – Comparison of Minimum Detectable Activity with Urinalyses Action Levels (Class Y Uranium; Method: Urinalysis; Minimum Frequency: Quarterly)

Level	Concentrations in Urine	
	(Bq L ⁻¹)	(pCi L ⁻¹)
Control	0.002	0.06
MDA	0.004	0.1
Work Restriction	0.007	0.2

Table 7 – Action Levels for Special Class Y Uranium^a
(Method: Urinalysis; Frequency: Quarterly)

Actions	Action Level	
	Concentrations in Urine	
	(Bq L ⁻¹)	(pCi L ⁻¹)
Evaluation	0.18	5
Control	0.55	15
Work Restriction ^b	1.83	50

^a See Appendix A.3.10.

^b If confirmed by at least three samples over a quarter.

Table 8 – Minimum Frequencies of Bioassay

Solubility Class	Situation	Frequency		
		Urine	Fecal	In vivo
D	Radiological			
	Routine	Monthly	^a	^a
	Special	<---	Daily or as needed	--->
W	Routine	Quarterly		Annually
	Special	<---	Daily or as needed	--->
Special Y	Routine	Quarterly		Annually
	Special	<---	Daily or as needed	--->
Y	Routine	^a	Quarterly	Annually
	Special	<---	Daily or as needed	--->
D and W	Chemical Toxicity			
	Routine	Monthly		Annually for Class W
	Special	<---	Daily or as needed	--->

^a The method of analysis not usually used.

Table 9 - Fractions of Inhaled Class Y Uranium in Fecal Compartment vs. Time^a

Day After Exposure	Fraction Excreted in Feces	
	For Day Listed	Cumulative Through Day Listed
1	0.052	0.052
2	0.160	0.212
3	0.131	0.343
4	0.074	0.417
5	0.036	0.453
6	0.017	0.470
7	0.008	0.478

^a From Lessard et al. (1987).

Appendix A

Rationale for Quantitative Values and Specific Provisions of This Standard

(This Appendix is not a part of American National Standard HPS N13.22-1995.)

Editor's Note: Although the use of scientific E-notation does not adhere to Health Physics Society style guidelines, it is used in this publication (Appendices) due to its citation in numerous tables, references, and text.

A.1 Toxicology Review and Rationale for Mass Levels of Uranium in Process Above Which A Bioassay Program is Needed (Section 3)

A.1.1 Uranium Toxicology Review

Establishing the need for bioassay programs for uranium requires consideration of radiation protection philosophy, the quantities of uranium that experience shows might be taken into the body relative to the amount in various processes, and the toxicity levels of uranium in its various nuclidic and chemical forms.

The management philosophy of most radiation protection programs is to maintain the exposures to each worker "as low as reasonably achievable (ALARA)," as well as to maintain exposures well within regulatory limits (U.S. NRC 1994). This philosophy is usually implemented by providing combinations of facility design, including exhaust ventilation and containment; equipment that reduces the time each worker is in proximity to possible sources of exposure; and procedures performed by well-trained workers (Brodsky 1980, 1989). In order to monitor whether the facility design and other safety provisions are functioning properly, equipment such as general air and breathing zone air monitors evaluate the air concentration of uranium to which workers might be exposed. These air monitors can be used to help determine if a worker might be exposed to uranium, such that he/she should be included in the bioassay program, and at what frequency. However, it is well demonstrated that air monitoring might not always detect, or adequately represent, the intakes of workers exposed to airborne uranium (Kenoyer et al. 1987). Thus, it is necessary to assess the quantities of uranium in process that might produce intakes that are measurable, that are significant relative to the regulatory and advisory limits on intake, or that approach levels at which investigatory and remedial steps should be taken to ensure that releases of uranium to air and the environment are not out of control.

Considerations of the chemical toxicity (for Class D and W compounds) require that, in addition to maintaining intakes of uranium well below regulatory annual or quarterly intake limits, operations must be designed to maintain exposures, including any single intakes, below levels that will cause transient kidney damage due to the toxicity of uranium.

In discussing the ICRP 6 chemical toxicity limit for soluble natural uranium (i.e., corresponding to a daily inhalation of not more than 2.5 mg), Eve (1964) pointed out that an acute inhalation of the 13-wk (65 workdays) radiological limit (U.S. NRC 1994; ICRP 1960, 1964) in a single workday would be equivalent to a single intake of 136 mg. She calculated that this intake would produce an initial urinary excretion level of 17 mg L⁻¹, and considered that this would be unjustifiably high compared to the level of 2 mg L⁻¹ at which albuminuria had been observed in some human accident cases. Thus, she ruled out a 13-week intake all in one exposure (or 1 d) and stated, "It would be reasonable therefore that 1 day's total exposure could be allowed in a single intake; this quantity is 2.1 mg in the air breathed...." (Eve 1964; Lawrence 1984). Thus, the ICRP 6 limit (ICRP 1964) for inhaled soluble natural uranium was actually based on a single day's inhalation at the radiological MPC_a (U.S. NRC 1994; ICRP 1960), with a large amount of conservatism built in. McGuire (1991) points out that a single intake of Class D uranium of any expected assay

equivalent to a 50 mSv (5 rem) dose would be an order of magnitude above this ICRP 6 limit and would be expected to cause kidney damage.

This ICRP 6 recommendation is conservative when compared with the summary of recent opinion of a panel of experts by Just and Emler (1984). McGuire has generally accepted the estimates of Just and Emler as appropriate for planning emergency response programs for NRC-licensed uranium processors (McGuire 1985). McGuire (1991) has also reviewed more recent data on human exposures (Fisher et al. 1990) and has summarized the effect for various intakes of uranium, as reproduced in Table A.1. These data indicate that a single intake of 8 mg of natural uranium would be well below the level causing permanent kidney damage in most individuals, and that 4 mg intake would probably cause no observable effects. This is not contradicted by the findings from the more recently reported Chinese cases (Lu and Zhao 1990), since the recommendations of Su Lu and Fu-Yao Zhao (1990) for lowering permissible amounts of uranium in the body are based on additional safety factors, and on only transient indications of kidney changes. The urine action levels for situations where chemical toxicity might be of concern are based on McGuire's interpretations (see Table A.1).

Recent Canadian guidance [and a meeting in 1992 of United States experts (J. N. Stannard, personal communication, July 1992)] has retained the $3 \mu\text{g g}^{-1}$ kidney limit as a basis for designing bioassay programs (Health and Welfare Canada 1987), as has the revised NRC 10 CFR Part 20. The Occupational Safety and Health Administration (OSHA) in the United States has recently selected a Permissible Exposure Limit (PEL) of 0.05 mg m^{-3} for soluble compounds, and 0.25 mg m^{-3} for insoluble compounds (U.S. Department of Labor 1989).

A.1.2 Selection of Uranium Bioassay Implementation Levels

Since all industries and laboratories in the United States will need to be able to demonstrate bioassay data to show compliance with regulatory limits, as well as good practice in minimizing exposures to toxic agents, it is appropriate to establish a "bioassay program implementation level" so that measurements can be made when average daily exposure levels might reach as much as 10% of the PEL. An evaluation of field and bioassay data indicates that, in the absence of local exhaust ventilation, chemical exposures to volatile or readily dispersible materials may be maintained below several percent of PEL when a worker handles or is exposed to a process having an uncontained amount in mass units below 10,000 PEL, where PEL is in mass units per cubic meter (Brodsky 1990). (This rule of thumb includes an additional safety factor of 100 to allow for possible toxicity or carcinogenicity of untested chemicals; carcinogenicity is *not* the limiting effect for uranium.) The PEL is a limit for daily application and will ensure detection of significant exposures resulting from accidental releases as well as daily exposures from routine operations. The OSHA PELs are generally as conservative as, and often consistent with, the Threshold Limit Value - Time Weighted Average (TLV-TWA) limits recommended by the American Conference of Governmental Industrial Hygienists (ACGIH) (1990); the TLV-TWAs for uranium are 0.2 mg m^{-3} for both soluble and insoluble compounds (U.S. Department of Labor 1989; ACGIH 1990). The OSHA PEL is 0.05 mg m^{-3} of uranium in air.

When uncontained soluble uranium is handled, the PEL formula above would indicate a minimum bioassay program might be needed at $10,000 \text{ m}^3 \times 0.05 \text{ mg m}^{-3} = 500 \text{ mg}$ (which is somewhat above the lethal quantity estimate in Table A.1). However, experience with natural uranium in industry shows that no more than a small fraction of such a quantity would be likely to escape from process vessels and enter the body of any worker. Considerations of the extremely high density and low specific activity of natural uranium have in the past been used – even for relatively dusty operations – to raise the quantities of uranium in process requiring bioassay by a factor of 100 (in relative ALI units) compared to most other radionuclides (Brodsky 1980, 1989). After consideration of its own broad industrial experience, the Working Group used a factor of 1,000 applied to the 500-mg value above to obtain the 0.5 kg quantity in Table 1 for natural uranium. (Note that in Section 3.2.1.3 a confirmatory bioassay program is

recommended at 0.1 of the levels in Table 1. Consider also that many of us have worked with one-pound bottles of uranyl nitrate on a chemical laboratory benchtop without significant internal exposure.) For processes conducted within a well-designed exhaust hood, this quantity can be raised by a factor of 10; for work within complete exhausted enclosures such as gloveboxes, an additional factor of 100 can be allowed. These factors are consistent with those recommended in various references (Brodsky 1980, 1989; Health and Welfare Canada 1987).

A recent British schema for determining maximum amounts of unsealed material that can be handled in laboratories with various categories of protection would allow up to 60 kg of uranium to be handled in a "Type C" medium quality chemical laboratory for normal operations; no recommendations regarding bioassay are given (Hudson and Shaw 1993). The British schema also takes into account the lower resuspension factors and lower release fractions of very low specific activity materials. However, in some operations, quantities of uranium in the kilogram range will require safety considerations such as criticality, in addition to internal exposure (Wagner 1994).

As a check, consider that the fractional intakes of material in process that have been observed to be taken into the bodies of workers (Brodsky 1980) are usually no more than 10^{-6} and almost always less than 10^{-5} by inhalation. These fractions may be applied to the 0.5-kg quantity in Table 1 to obtain intakes of no more than about 0.5 - 5 mg, a range below the level of observed toxicity. Also, the upper levels of fractional intake include a variety of processes and accident conditions in which the primary laboratory container was breached by explosive forces (Brodsky 1980).

In the case of uranium mills, with large quantities of uranium in very dusty processes, some of which can leak even when designed for a certain degree of containment, many operations can be assumed to require at least a minimum bioassay program together with regular air monitoring [see Regulatory Guide 8.22 (U.S. NRC 1988)].

A.2 Radiological Toxicity and Rationale for Bioassay Program Implementation Levels Based on Activity for Enriched Uranium (Section 3)

The assessment of radiological toxicity in this standard is based on the recommendations of the International Commission on Radiological Protection (ICRP) (1979), which are the bases for regulatory limits of the U.S. Nuclear Regulatory Commission (1991). These ICRP recommendations on Derived Air Concentrations (DAC) for continuous exposure differ from the earlier Maximum Permissible Concentrations (MPC) (ICRP 1960) by only factors of two or three for most radionuclides (Brodsky 1989).

However, the limits for some uranium compounds have been lowered by factors of about six (see Table A.2). Also, the ICRP 30 recommendations of Annual Limits on Intake (ALIs) allow the rates of intake to vary as long as ALIs are not exceeded in any 1 y.

Using the most restrictive DAC data in becquerels per cubic meter from Table A.2, 0.6 Bq m^{-3} , and multiplying values by 10,000 DAC (in units of m^3) (as for 10,000 PEL in the case of chemical toxicity) yields 6,000 Bq. Multiplying by the additional factor of 1,000 to take into account the lower probability of intake of uranium in process -- as in Section A.1 -- the implementation level of Table 1 for bioassay of a worker handling unsealed enriched uranium becomes $6 \times 10^6 \text{ Bq}$. Table A.3 shows the transportability classes of several compounds of uranium, which shows that many compounds will have higher DACs than 0.6 Bq m^{-3} .

A cautionary note regarding chemical toxicity must also be added here for enriched uranium. McGuire (1991) has shown that, even for high enrichments of uranium in the ^{235}U isotope, a single intake in activity units to give a 0.25-Sv (25-rem) committed effective dose equivalent (ICRP 1979) could exceed the OSHA

PEL based on chemical toxicity. The Canadian guidance (Health and Welfare Canada 1987), on the other hand, suggests that chemical toxicity is the dominant consideration over radiological toxicity only for the more transportable Class D compounds, under conditions of equilibrium after constant intake at 0.3ALI/365 per day (see Table A.4). However, if a lower kidney burden than 900 mg uranium by a factor of four were established, as for the OSHA PEL, a higher specific activity than that of natural uranium (2.5×10^7 Bq kg⁻¹) could bring the equilibrium mass burden equivalent to 6.5 Bq above the lowered permissible chemical burden (i.e., 900/4 mg in the total kidney). Then, under continuous exposure conditions, both Class D and W natural uranium could exceed permissible chemical levels in the body after some months of continuous exposure at 0.3 DAC levels.

Appendix C of the Canadian guidance also calculates that below an enrichment of 20% ²³⁵U by weight, a 900-mg kidney burden could be exceeded in a single intake without exceeding the ALI. Thus, chemical toxicity is suggested as the primary concern for single intakes of Class D material in a routine monitoring program, with a requirement to consider chemical toxicity for Class W for depleted uranium also. It is shown that, for a 900-mg kidney limit, radiological considerations limit the Derived Investigation Levels (DILs) for enrichments greater than 20%, although in the long-term the possible chemical toxicity must be considered also even for these higher enrichments.

Thus, although the Working Group believes that the action levels and actions in this standard are already conservative enough for worker protection, with the more conservative regulatory approach prevailing in the United States regarding chemical toxicity, bioassay programs recommended in this standard for enriched uranium might also require some degree of surveillance over chemical toxicity (Thompson 1994). This surveillance can be of a confirmatory nature using determinations of mass quantities of uranium excreted from a sample of employees, as appropriate to the particular operations and mass concentrations determined in air. No further specific recommendations for monitoring enriched uranium processes were deemed appropriate for this standard.

A.3 Rationale for Action Levels of Section 6

A.3.1 Action levels were developed based on the approach in ICRP 54 (1988). This approach designates derived levels (DLs) of radioactivity in bioassay samples that would be expected for single intakes at the middle of the sampling interval that, if continued on the average for each interval, would result in 0.1 ALI, 0.3 ALI, and 1 ALI in 1 y (see relationship in Section 6.1). The DLs in ICRP 54 have been designated as "Recording Level" (RL), "Investigation Level" (IL), and "Action Level" (AL), respectively. In this uranium standard, the ICRP 54 derived levels for RL, IL, and AL have been replaced by more conservative action-level designations: the "Evaluation Level (EL)" is taken to be the same numerically as the ICRP 54 RL; the "Control Level (CL)" is numerically equal to the ICRP 54 IL; and the "Work Restriction Level (WRL)" is equal to the ICRP 54 AL, respectively. Current regulations, policies, and practices in the United States are more conservative than those generally outlined in ICRP 54. Also, as summarized in Appendices A.1 and A.2, there is some evidence of possible toxic effects of uranium at low levels that might not be discernible from routine diagnostic data. The actions corresponding to the EL, CL, and WRL of this standard are a part of the standard and thus are defined in Sections 6.1.3 and 6.1.4.

Further, this standard recommends recording all bioassay data and maintaining them in a permanent file. This is required by applicable regulations in the United States. Recording and reporting provisions are also included in other standards (Health Physics Society 1987; ANSI 1972). Thus, the recordkeeping and reporting provisions of this standard have been kept brief; they are included for completeness and to remind the reader that proper records and reports, as defined by other standards, are a part of an acceptable internal dosimetry program. Other useful bioassay program recommendations can be found in ICRP and NCRP reports listed in the Bibliography found in Section E.2.

A.3.2 The action levels are based on the retention and excretion parameters assumed by the ICRP for uranium particles with an AMAD of 1 μm (action levels derived using this assumption are conservative) in the interest of safety. If the experience at a facility shows different retention and excretion factors or a different AMAD, the action points can be adjusted accordingly.

A.3.3 The action levels of Section 6.1, Tables 2 through 8, are for employees who have had *no* previous exposure that still results in positive bioassay samples. If employees have had previous exposure, these action levels might need to be adjusted upward to take into account the amount of activity in the urine or lung, and its day-to-day variations, which are attributable to previous exposure. The detection of additional exposures from current operations would then be limited by an MDA calculated using the total variation, s_p , including variations in bioassay results from previous exposure (see Appendix D).

A.3.4 The action levels listed in this standard are for the long-lived isotopes of uranium (^{238}U , ^{236}U , ^{235}U , ^{234}U , and ^{233}U). As noted in the report of the Canadian Working Group on Bioassay (ICRP 1988), dosimetry factors for the uranium isotopes with which we are dealing in this document are within +10% to -5% of the dosimetry factors for ^{235}U that were used in the determination of the action levels shown. In view of this fact it was decided that it was unnecessary to have separate action levels for different isotopes. Facilities can make adjustments for the isotopic mixture(s) they process as the need arises.

A.3.5 However, if the air contamination level is consistently or frequently elevated, it could be necessary to adjust the action points to better help assure that exposure control is optimized. In addition, more frequent monitoring may be required after an exposure accident or incident occurs, or when very high acute exposures must be detected within certain times to allow effective medical intervention.

A.3.6 The first action levels of Table 2 are based on radiological considerations. For Class D and W materials, the mass amount of uranium may exceed the safe chemical toxicity level before the radiological toxicity levels are reached. A urine limit of just over 116 mg d^{-1} can be calculated using (1) McGuire's table (Table A.1), which he developed from Just and Emler to determine the intake that is the threshold for transient renal injury or effect, i.e., 8 mg; (2) results from ICRP 54 (1988) as presented in Fig. A.1. and A.2. which show that, according to the ICRP, for acute exposures the ratio of the amount of uranium in the kidney to the amount in the urine did not exceed four during the first hundred days after an acute exposure and did not exceed three at anytime during chronic exposure; and (3) ICRP model calculations indicating that the maximum activity in the kidney is 5.83% of intake. Adjusting for an excretion rate of 1.4 L d^{-1} , this equates to about 85 $\mu\text{g L}^{-1}$ for adoption as the work restriction level. The evaluation levels and the control levels were then set to 0.1 and 0.3 of this figure for consistency with the other action points. After rounding, this gives action points as shown in Table 2. Such action levels should help assure that transient renal effects are improbable, and permanent renal damage could not result unless the removal action point (WRL) is greatly exceeded.

The implementation of special urine sampling when intake could have exceeded 2 mg Class D or 8 mg Class W uranium in a day (Section 6.4) should further reduce the probability of any transient renal effect.

A.3.7 Uranium Class Y materials cannot be effectively detected at the levels listed in ICRP Publication 54 by the ordinary methods available for either lung *in-vivo* counts or urinalyses.

This is shown by the fact that the DIL (designated as the control action level in this standard) was 0.002 Bq L^{-1} (0.06 pCi L^{-1}) (Table 6), which is below the MDA suggested as reasonable for routine uranium alpha

urinalysis in the Health Physics Society standard (0.1 pCi L^{-1}) (Health Physics Society 1987). However, Singh (1994), who presents an investigation level (IL) of 0.0023 Bq for daily urinary excretion of Class Y material [close to the control level (CL) of Table 6 of this standard] and a detection limit of 0.01 Bq L^{-1} by alpha spectrometry, cites other but perhaps more expensive analytical methods for determining urinary excretion of Class Y material at the action levels of this standard.

An approach assuming a special Class Y material has been taken in Section 6 to provide the action levels of Table 7, which would be applicable to many facilities (see Section A.3.10).

A.3.8 Although routine fecal analysis programs are sometimes considered to have esthetic and resultant administrative difficulties, fecal action levels were also calculated for Class Y material using ICRP 54 methodology and are tabulated in Table 3, since fecal analysis is often more likely than urinalysis to detect exposure to very refractory (highly insoluble) Type Y material. It is noted that the ratio between the fecal excretion level per day and the urine excretion level per liter is greater than seven as calculated for this 90-d sampling interval and that *all* action levels are above the attainable MDA specified for fecal analysis of 0.1 pCi per sample (3.7 mBq per sample) (Health Physics Society 1987). It is apparent from the above information that facilities that have significant Class Y uranium exposure potential should have fecal analysis capabilities available to them (unless they have urinalysis methods that have MDAs well below 0.1 pCi per sample) (3.7 mBq per sample).

A.3.9 As stated in Section 6.7, the action (CL) levels of Table 4 for *in-vivo* lung counting are often below the MDAs listed for comparison. *In-vivo* lung counting varies in sensitivity depending on the isotopic mixture involved. Uranium lung *in-vivo* monitoring is accomplished by detection of the 186-keV gamma ray from ^{235}U , or the 93- and 63-keV x rays from a ^{238}U decay product (assuming equilibrium). This information is used to estimate the total activity based on the amount of ^{235}U or ^{238}U activity and knowledge of, or assumptions about, the uranium isotopic mixture involved.

Furthermore, these two isotopes do not exist in the pure state but rather in combination with ^{234}U in varying amounts. The weight percent of ^{234}U is increased in the enrichment process as the percent of ^{235}U is increased. This increase results in most of the radiation dose from such mixtures of enriched uranium coming from the ^{234}U isotope, which cannot be measured in quantities of interest by present *in-vivo* methods. Table 4 shows the total activity at various levels of ^{235}U and ^{238}U , illustrates what *in-vivo* lung counters can detect, and how far these detection levels are above 75 Bq , which is the control level (CL) for ^{235}U for Class Y uranium with an annual monitoring frequency. (Note that the ICRP's DIL and the "control" level used in this standard are the same.) [The specific activities used for these calculations are from the formula reported by Alexander (1974). Possible deviations from these specific activities should be noted (Manninen 1994).]

This point is further illustrated and emphasized in Table A.5, which illustrates increasing total uranium activity vs. isotopic enrichment, which may be compared with the control (CL) and work restriction (WRL) levels developed by the techniques used here and the MDA performance criteria in Table 4 for *in-vivo* monitoring. Of course, for a given lung counting system and procedure, the MDA can be lowered (improved) inversely as the square-root of the counting time (see Section 9.4) or by averaging a larger number of determinations (Strom and McGuire 1993).

The reasonably attainable MDAs for ^{235}U and ^{238}U are 7.4 Bq (0.2 nCi) and 110 Bq (3 nCi), respectively, for typical bioassay *in-vivo* laboratories (Health Physics Society 1987; Brodsky 1986; Crawford-Brown and Wilson 1984). However, the best state-of-the-art techniques have attained MDAs of 4 Bq (0.12 nCi) and 70 Bq (1.8 nCi), respectively (Palmer et al. 1987; Toohey et al. 1991). A large percent of the total activity quoted above comes from the associated ^{234}U .

Note: The information given in Table A.5 is *only* an example for two specific mixtures of uranium isotopes. The total activity from mixtures will vary at the MDAs for ^{235}U and ^{238}U depending in large part on how much ^{234}U is in the mixture (Manninen 1994). Consequently, each facility will need to consider this fact in establishing the most appropriate monitoring technique and in determining the MDA (for total activity) for the particular blend(s) of isotopes it handles.

A.3.10 Special Class Y Action Levels

In Section 6.5 and Table 7, the working group provided action levels for a "Special Class Y" material, with an assumed 100-d effective half-life in the lung, and excretion parameters the same as Class W material. This was based on experience in certain laboratories that showed some uranium oxides and compounds are solubilized in body fluids much faster than assumed in the ICRP 30 report. The behavior of this special Class Y material is similar to that more recently reported by Forrest and Barber as "Class Q" (Forrest and Barber 1993). In ICRP 30, uranium oxides are characterized as Class Y material, with a 500-d clearance time in the pulmonary region. Also, ICRP 30 calculations assume a 1-micron particle size. An 8-micron size was obtained for Class Q material obtained from plant processes, and Class Q was found to be composed of two parts: 90% Class W material with a 120-d pulmonary half-life clearance time and 10% Class Y material with a 500-d pulmonary half clearance time.

Forrest and Barber (1993) calculated an Annual Limit on Intake (ALI) of 23,400 Bq for this Class Q material. This ALI for Class Q material may be compared with 2,000, 30,000, and 50,000 Bq for Class Y, W, and D ^{235}U , respectively, and 2,000, 50,000, and 50,000 Bq for Class Y, W, and D ^{238}U , respectively (ICRP 1979). Thus, Class Q material would not be expected to differ much in excretion rates or radiotoxicity from the special Class Y material defined in this standard. In any case, Section 6.5 cautions that the relative transportability of the material should be determined and documented before the action levels for special Class Y material are used. Otherwise, other available distribution and excretion models (Thompson 1994; Forrest and Barber 1993; Fisher and Briant 1994; Singh 1994; Wrenn and Bertelli 1994; Bernard 1977) should be used to determine appropriate action levels, or the more conservative (on the safe side) action levels for the particular situation, as given in this standard, should be used.

A.3.11 Limitations of the "Missed Dose" Concept

A "missed dose" is often considered that magnitude of internal dose commitment that would be received if a bioassay sample at the end of a sampling interval contained a "minimum detectable amount (MDA)" of radionuclide after an intake immediately following the previous sampling. However, the fact remains that by the accepted definition of MDA (Brodsky 1986), there would be only an 0.05 chance of calling the sample "negative." For example, if an employee were regularly to take in a quantity, just after each monthly bioassay, that would yield an MDA quantity in the urine at the next bioassay interval, even if we neglected the remaining amount from the previous intake the probability that such an exposure situation would be missed for all 12 mo of the year would be only $(0.05)^{12} = 2.4 \times 10^{-16}$, which is a negligible probability. Furthermore, the assumption that all intakes occur only immediately after each sampling is over-conservative. The more likely scenario is that exposures will occur randomly over time, with equal probability per unit time interval when process events are unrelated to bioassay sampling times. Also, one should certainly not accumulate such values of intake corresponding to MDAs at sampling time on a monthly basis and then assume the total is the missed annual dose.

A.3.12 Uranium Mill Action Levels

As indicated in Section 6.8, uranium mill action levels are incorporated in this standard only by reference to Regulatory Guide 8.22 of the U.S. Nuclear Regulatory Commission (1988). However, the action levels in Regulatory Guide 8.22 are not greatly different from those in this standard. The action levels for mills were derived with a model that does not consider the bone compartment separately for long-term build-up of Class Y material (Alexander et al. 1986). However, the bone doses would be small compared to kidney doses for the types of exposure encountered in the mills (Kathren and Weber 1988; Chase 1988), and

bone doses within the limits of regulatory standards would be well below any observed thresholds for the induction of bone cancer or leukemia (Thomas 1994).

Table A.1 – Health Effects From Acute Intake of Soluble Uranium (from McGuire 1991)

Health Effects	Uranium per kg body wt. (mg U kg ⁻¹) ^a	Uranium in 70 kg person (mg)	Uranium intake by 70 kg person (mg) ^b
50% lethality	1.63	114	230
Threshold for permanent renal damage	0.3 ^c	21	40
Threshold for transient renal injury or effect	0.058	4.06	8
No effect	0.03	2.1	4

^a Based on review of Just and Emler (1984), except as noted.

^b Intake is defined as the total amount of material inhaled into the body. It includes material immediately exhaled in addition to material absorbed within the body. For small uranium particles in soluble form, about half the intake will be absorbed by the body according to ICRP 30 (1979).

^c Based on the conclusions of Wrenn (see discussion in Just and Emler 1984).

Table A.2 – Comparison of ICRP 2 MPCs, ICRP 30 DACs, and ALIs

Isotope	ICRP 30 LEVELS			ICRP 2 MPC Levels		
	ICRP 30 Class ^c	ALI (Bq)	DAC (Bq m ⁻³)	DAC (μCi cm ⁻³)	ICRP 2 Class	40 h wk ⁻¹ (μCi cc ⁻¹)
²³⁴ U ^a	D	5E(4)	2E(1)	5.4E(-10)	S	6E(-10)
	W	3E(4)	1E(1)	2.7E(-10)		
	Y	1E(3)	6E(-1)	1.6E(-11)	I	1E(-10)
²³⁵ U ^a	D	5E(4)	2E(1)	5.4E(-10)	S	5E(-10)
	W	3E(4)	1E(1)	2.7E(-10)		
	Y	2E(3)	6E(-1)	1.6E(-11)	I	1E(-10)
²³⁸ U ^a	D	5E(4)	2E(1)	5.4E(-10)	S	7E(-11)
	W	3E(4)	1E(1)	2.7E(-10)		
	Y	2E(3)	7E(-1)	1.9E(-11)	I	1E(-10)
U-NAT	Not Listed. Must Be Calculated For Specified Mix.				S	7E(-11)
					I	6E(-11)

^a This footnote is quoted from footnote 4 in the 10 CFR Part 20, effective before the revision (U.S. NRC 1991) since it might be useful for interpreting results for some forms of uranium:

"If a mixture of radionuclides consists of uranium and its daughters in ore dust prior to chemical separation of the uranium from the ore, the values specified below may be used for uranium and its daughters through ²²⁶Ra, instead of those from paragraphs 1, 2, or 3 above.

1. For purposes of Table I, Col. 1 – 1 E(-10) μCi mL⁻¹ gross alpha activity; or 5 E(-11) μCi mL⁻¹ natural uranium; or 75 μg m⁻³ of air natural uranium.
2. For purposes of Table II, Col. 1 – 3 E(-12) μCi mL⁻¹ gross alpha activity; 2 E(-12) μCi mL⁻¹ natural uranium; or 3 μg m⁻³ of air natural uranium."

"If a mixture of radionuclides consists of uranium and its daughters in ore dust (10 μm AMAD particle distribution assumed) prior to chemical separation of the uranium from the ore, the following values may be used for the DAC of the mixture: 6 (E-11) μCi of gross alpha activity from uranium-238, uranium-234, thorium-230, and radium-226 (*sic*) per milliliter of air; 3 (E-11) μCi of natural uranium per milliliter of air; or 45 micrograms of natural uranium per cubic meter of air."

^b ICRP 2 does not give MPCs for the public but does give them for workers on a 168-h wk⁻¹ basis. It is these 168-h wk⁻¹ values that were originally divided by 10 to obtain MPCs in Table II of 10 CFR Part 20 for exposures to persons in unrestricted areas (the public).

ICRP 30 gives no values for the public, only for workers. Values were presented in the proposed revision to 10 CFR Part 20 (U.S. NRC 1991) as "reference levels" that would ensure exposures to members of the public remained below 0.1 rem y⁻¹ equivalent.

Footnotes to Table A.2 (continued)

^c In ICRP 30, Class D compounds include UF₆, UO₂F₂, and UO₂(NO₃)₂; Class W includes UO₃, UF₄, UCl₄; and Class Y includes UO₂ and U₃O₈. See Table A.3.

Other information needed for calculating action levels and dose assessment for different enrichments of uranium are contained in Regulatory Guide 8.22 and 10 CFR Part 20.

A footnote to Appendix B of 10 CFR Part 20 (U.S. NRC 1994, p. 290), which is the same in the proposed revision (U.S. NRC 1991), provides the relationship between specific activity and enrichment of ²³⁵U in the uranium:

"For soluble mixtures of ²³⁸U, ²³⁴U, and ²³⁵U in air chemical toxicity may be the limiting factor. If the percent by weight enrichment of ²³⁵U is less than 5, the concentration value for a 40-hour workweek, Table I, is 0.2 milligrams uranium per cubic meter of air average (*sic*). For any enrichment, the product of the average concentration and time of exposure during a 40-hour workweek shall not exceed 8 E(-3) SA μCi h mL⁻¹, where SA is the specific activity of the uranium inhaled. The concentration value for Table II is 0.007 milligrams uranium per cubic meter of air. The specific activity for natural uranium is (25,000 Bq (6.77 (E-7) curies) per gram U. The specific activity for other mixtures of ²³⁸U, ²³⁵U, and ²³⁴U, if not known, shall be:

$$SA = 3.6 E(-7) \text{ Ci/g U } \quad \text{U depleted} \quad (\text{A.1})$$

$$SA = (0.4 + 0.38 E + 0.0034 E^2) (E-6) \quad E \geq 0.72 \quad (\text{A.2})$$

where E is the percentage by weight of ²³⁵U, expressed as percent."

If the fractional amount of mass of each isotope, ²³⁴U, ²³⁵U and ²³⁸U is known, and they are f₄, f₅, and f₈, respectively, and if the specific activities of the pure nuclides ²³⁴U, ²³⁵U and ²³⁸U are SA₄, SA₅ and SA₈, respectively, then the exact specific activity of the uranium mixture can be calculated as:

$$SA_{\text{mix}} = f_4 SA_4 + f_5 SA_5 + f_8 SA_8 \quad \text{Ci/g} \quad (\text{A.3})$$

A convenient equation for determining specific activity of a pure nuclide is (Cember 1983):

$$SA_i = A_{\text{Ra-226}} T_{\text{Ra-226}} / A_i T_i \quad \text{Ci/g} \quad (\text{A.4})$$

or, if SI units are used,

$$SA_{i(\text{SI})} = 3.7 \times 10^{13} (A_{\text{Ra-226}} T_{\text{Ra-226}} / A_i T_i) \quad \text{Bq/g} \quad (\text{A.5})$$

where the mass number of the ²²⁶Ra atom is A_{Ra-226} = 226 and the half-life of ²²⁶Ra, T_{Ra-226} = 1600 y (Brodsky 1979; Kocher 1981). Since the half-lives and mass numbers for ²³⁴U, ²³⁵U, and ²³⁸U are, respectively, 2.445 x 10⁵ y, 7.038 x 10⁸ y, and 4.468 x 10⁹ y, and 234, 235 and 238, the specific activities of these nuclides are:

$$SA_{234} = (226)(1600)/(234)(2.445 \times 10^5) = 6.320 \times 10^{-3} \text{ Ci/g} \quad (\text{A.6})$$

$$SA_{235} = (226)(1600)/(235)(7.038 \times 10^8) = 2.186 \times 10^{-6} \text{ Ci/g} \quad (\text{A.7})$$

Footnotes to Table A.2 (continued)

$$SA_{238} = (226)(1600)/(238)(4.468 \times 10^9) = 3.400 \times 10^{-7} \text{ Ci g}^{-1} \quad (\text{A.8})$$

For example, we may use these values to calculate the specific activity of natural uranium, assuming 0.711% ^{235}U , and assuming that the ^{234}U is in secular equilibrium with the ^{238}U :

Since the specific activity of ^{234}U is four orders of magnitude higher than for ^{238}U (eqns A.6 and A.8), the mass of ^{234}U may be neglected. One gram of natural uranium may then be taken to be composed of 0.711 (E-2) g of ^{235}U and 0.99289 g ^{238}U . The specific activity of the mixture is then $(0.99289)(3.400+3.400)(\text{E}-7) + (0.711 \text{ E}-2)(2.186 \text{ E}-6) = 6.7516 \text{ E}-7 + 0.00155 \text{ E}-7 = 6.7532 (\text{E}-7)$, close to the value of $6.77 \text{ E}(-7)$ in 10 CFR Part 20 for the specific activity of natural uranium (probably based on earlier data). These values agree within less than 0.3 % so either is more than sufficiently accurate for bioassay purposes.

From the footnote on page 2 of Regulatory Guide 8.22, which has been checked against recent nuclear data:

"The $1 \text{ E}(-10) \mu\text{Ci mL}^{-1}$ ($3.7 \text{ E}(-6) \text{ Bq/mL}$) value is not exactly consistent with the 0.2 mg/m^3 concentration limit for soluble uranium in Footnote 4 of Appendix B to 10 CFR Part 20 because of the rounding off of values in Appendix B. Since the $1 \text{ E}(-10) \mu\text{Ci mL}^{-1}$ limit is more restrictive, this value has been used in the calculation of all the action levels (weekly and quarterly) in this guide. For compliance purposes, Footnote 4 to Appendix B sets the weekly limit for soluble uranium compounds, which can be converted to radiological units using the specific activity of natural uranium ($6.77 \text{ E}(-7) \text{ Ci/g}$ or $2.4 \text{ E}(4) \text{ Bq/g}$). As now defined in 10 CFR Part 20, the curie of natural uranium differs from the original definition in ICRP-2....The present definition of the curie of natural uranium in 10 CFR Part 20 refers to the total activity of all uranium isotopes in the natural mixture. When natural uranium is defined to be 0.711% by weight ^{235}U and the ^{234}U is assumed to be in secular equilibrium with ^{238}U , 1 Ci of natural uranium is composed of 0.489 Ci ^{234}U , 0.0225 Ci ^{235}U , and 0.489 Ci ^{238}U . Actual percentages of ^{235}U may be $0.711 \pm 0.1\%$."

By contrast, the "special curie," which was used in earlier literature and in the paper by Cofield (1966), was defined as $3.7 \text{ E}(10)$ disintegrations per second (dps or Bq) of ^{238}U , $3.7 \text{ E}(10)$ dps of ^{234}U , and $9.0 \text{ E}(8)$ dps of ^{235}U , yielding 7.49 alpha dps/Ci, and an MPBB of natural uranium of 749 alpha dps for insoluble uranium (Cofield 1966). [The maximum permissible body burden (MPBB) for natural uranium in ICRP-2 is only $5 \text{ E}(-3) \mu\text{Ci}$ due to consideration of chemical toxicity in ICRP 2 (ICRP 1960, p.81), even though the long-term fraction of activity in kidney is taken as only 0.065 compared to 0.85 in bone, and the biological half-lives are taken as 15 d and 300 d, respectively.] Since each isotope has a total alpha abundance close to 1 per disintegration (Kocher 1981), the ratios of activity given by Cofield normalized to 1 Ci of activity give: 0.494 Ci ^{234}U , 0.494 Ci ^{238}U , and 0.012 Ci ^{235}U ; this does not agree with the ratios given in Regulatory Guide 8.22, so care is needed in using some of the older literature in calibration of detector systems.

Also, variations in ^{235}U content of "natural" uranium mixtures can occur. Using the original ICRP 2 definition of the curie, and the ratios given in Regulatory Guide 8.22, the MPBB of $0.005 \mu\text{Ci}$ (in the total body when the kidney is at its limit) becomes $0.005 \times 2.22 \text{ E}(6) \text{ d/min-}\mu\text{Ci} / 60 \text{ sec/min} = 185 \text{ dps } ^{238}\text{U}$, $185 \text{ dps } ^{234}\text{U}$ and $8.5 \text{ dps } ^{235}\text{U}$, which totals 378.5 dps (Bq) for soluble natural uranium.

The value of 749 dps of Cofield (1966) for MPBB corresponds to the $0.01 \mu\text{Ci}$ he calculates for insoluble uranium when the lung is receiving its permissible dose rate (ICRP-2) of 0.3 rem per week (ICRP 1960). This alpha activity correspondence to dose rate can be used in emergencies to estimate approximate lung dose rates and intakes from *in-vivo* measurements, using the

gamma-ray abundances given earlier. For soluble uranium, the MPBB of 0.005 μCi in units of the "old" curie would be equivalent to about 0.01 μCi in units of the "new" curie, and in mass units would be $0.01/6.77 \text{ E}(-7) \mu\text{Ci per } \mu\text{g} = 1.48 \text{ E}(4) \mu\text{g} = 14.8 \text{ mg}$. For ICRP 2, the fraction of uranium in the kidney of that in the whole body is 0.065, which gives a maximum permissible continuous kidney burden of $0.065 \times 14.8 \text{ mg} = 962 \mu\text{g}$. This value is in reasonable agreement with the limit of 900 μg , quoted by Alexander et al. (1986), considering the rounding of numbers in the ICRP 2 report.

Cautionary Note: Recent determinations of specific activity at the Portsmouth Gaseous Diffusion Plant have shown that variations of about a factor of two from the specific activity formulae in 10 CFR Part 20 can be obtained, but these variations are limited to variations of about 30% for enrichments used in commercial fuel (Manninen 1994). Variations are due mainly to variations in the ^{234}U to ^{238}U ratio for different gaseous diffusion processes.

Table A.3 – GI Tract Absorption Factors For Inhalation Classes of Uranium (ICRP 1979, 1988)

Inhalation Class	Chemical Form	Absorption Factor (f ₁)
D	UF ₆ , UO ₂ F ₂ , UO ₂ (NO ₃) ₂	0.05
W	UO ₃ , UF ₄ , UCl ₄	0.05
Y	UO ₂ , U ₃ O ₈	0.002

Note: Some compounds that have been classed as Y have in humans shown more rapid clearance from lung than the 500-d half-life, often more like a 100-d clearance or less. This is perhaps due to mixtures of more than one physico-chemical form of material (ICRP 1988; Fisher et al. 1990). See Table 7. f₁ = fraction absorbed from GI tract into the blood (i.e., the fraction of a stable element reaching the body fluids following ingestion).

Table A.4 – Specific Activity of Uranium Corresponding to Given Equilibrium Radiological Burdens, in Order to Have 900 μg Uranium in Kidney (Health and Welfare Canada 1987)

Class	Equilibrium Kidney Burden from Continuous Intake at Rate of 0.3 ALI/365 d ⁻¹	Specific Activity of Uranium Mixture to Give Uranium Mass of 900 μg in Kidney (Bq kg ⁻¹)
D	43 Bq	4.8×10^7
W	6.5 Bq	7.2×10^6 ^a
Y	0.14 Bq	1.6×10^5 ^a

^a There are no uranium isotopes with a specific activity this low. Therefore, under the specified condition, a kidney burden of more than 900 μg could *not* occur. However, this fact does not mean that the 900 μg in the kidney could not be exceeded with a continuous intake of Class W material between 0.3 ALI/365 d⁻¹ and 1.0 ALI/365 d⁻¹, or in an acute exposure situation. It is assumed that the ^{234}U content is the amount that, along with the content of ^{235}U and ^{238}U , will give the listed specific activity.

Table A.5 – Uranium Activity Versus Isotopic Enrichments

Percent ²³⁵ U	Bq Total Uranium Activity for 7.4 Bq of ²³⁵ U (HPS 1987) ^a
3	180
5	163
7	158
10	156
15	159
20	161
30	170
40	180
50	192
60	203
70	214
80	226
93	240

Percent of ²³⁵ U	Percent of ²³⁸ U	Bq Total Uranium Activity for 110 Bq ²³⁸ U (HPS 1987) ^a
0.0	100.0	110
0.2	99.8	143
0.7	99.3	228
1.0	99.0	279

^a The values of 7.4 Bq (0.2 nCi) ²³⁵U and 110 Bq (3 nCi) ²³⁸U were estimated to be reasonably attainable MDAs for *in-vivo* counting (Health Physics Society 1987). See cautionary note regarding relative isotopic percentages at the end of the footnote to Table A.2.

Fig. A.1 – Ratio of Uranium in Kidney-to-Urine Concentration (Acute Inhalation) (from ICRP 54, page 6-63)

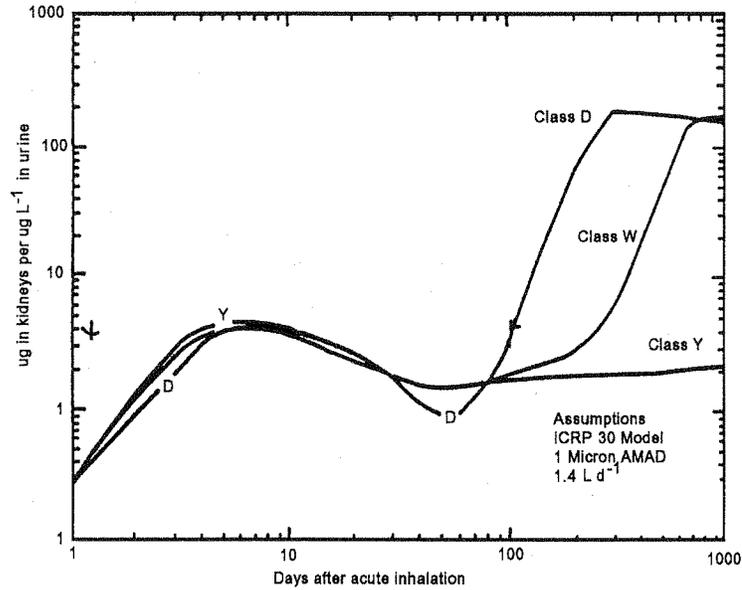
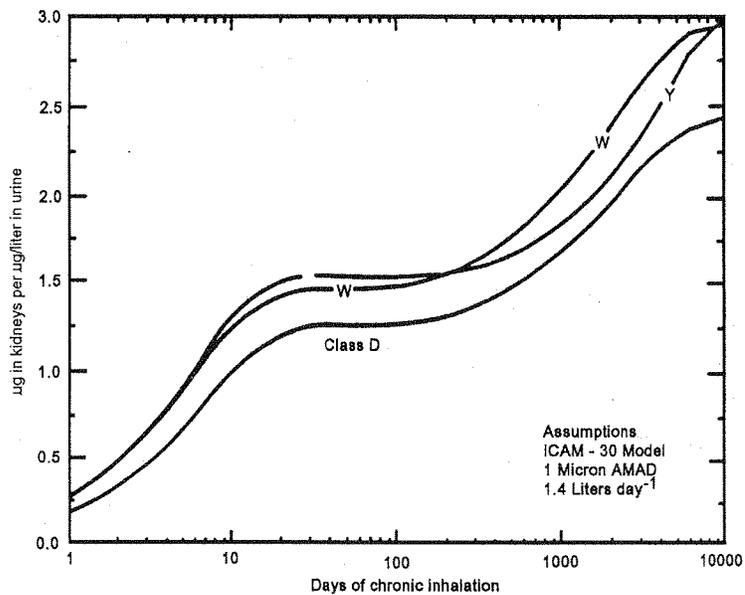


Fig. A.2 – Ratio of Uranium in Kidney-to-Urine Concentration (Chronic Inhalation) (from ICRP 54, page 6-65)



Appendix B

Evaluation of Intakes and Internal Burdens of Uranium and Organ Doses

(This Appendix is not a part of American National Standard HPS N13.22-1995.)

B.1 Simplified Methods for Estimating Intakes

All bioassay data should be examined by the plant health physicist and evaluated in timely fashion to provide optimum protection of workers and meet all regulatory reporting requirements. The following sections provide simplified and conservative analyses that may be used to ensure early recognition, early mitigatory action, and timely reports as applicable to workers, physicians, worker supervisors, plant management, and appropriate regulatory authorities. Also, the simplified methods in this section may be used for routine dose interpretation when excretion levels are below action levels requiring worker restriction and/or long-term follow-up.

For higher exposures, long-term follow-up is appropriate to improve estimates of intake and dose consistent with the possible need for long-term medical evaluation and for assessing the need for early and/or continuing decorporation therapy. Appropriate models can be fitted to sequential data taken over time on each individual to obtain sequential improvements in intake and dose assessments for these purposes.

B.1.1 Intake Estimates of Class D and W Exposure by Simple Computation From Urinary Data for Low-level Chronic Exposure (not exceeding 8 DAC-hours in Any 1 d)

The assessments will assume equilibrium between average deposition rate and average urinary excretion rate, and will be made according to the following procedure. (Units will be in mg L^{-1} and mg d^{-1} for illustration, but any set of consistent units may be used.)

1. Determine average urinary excretion concentration, C , over a calendar quarter, in $\mu\text{g U/L}$.
2. Assume 1.4 L excreted per day for each worker.
3. Multiply C by 1.4 to get the average daily excretion rate $E = 1.4 C$, in $\mu\text{g d}^{-1}$.
4. Multiply E by 2 to get daily intake by inhalation, $DI = 2.8 C$ in $\mu\text{g d}^{-1}$.

(This assumes that the excretion rate equals the deposition rate and assumes that half of the uranium inhaled in each breath is exhaled, as in the ICRP 2 simplified model (ICRP 1960). The lung model used in ICRP 30 will provide a result not more than about 20% different (ICRP 1979) for 1 micron AMAD particles or mists; 1 micron AMAD is a sufficiently conservative assumption. Individual variations in fractional rates of uptake of any transportable class material will vary more than these variations in uptake between models.)

5. Multiply DI by 65 workdays per quarter to get estimated maximum quarter's intake, $QI = 65 \times 2.8 C = 182 C$. This is likely to be conservative for workers monitored weekly when no air sampling or other information indicates any single releases that produce large increases in breathing zone concentrations. (A test of this hand computational method against a computerized calculation using ICRP 30 data follows: Just and Emler (1984) obtain $1.5 \mu\text{g L}^{-1}$ urine concentration for a chronic ^{238}U intake of 5 Bq y^{-1} . Using $182C$, the quarterly intake would be $182 (1.5) = 273 \mu\text{g}$ per

quarter, $1092 \mu\text{g y}^{-1} = 1092/82 \mu\text{g Bq}^{-1} = 13.3 \text{ Bq}$, less than three times the 5 Bq in the ICRP 30 calculation. The simplified calculation is shown here to be conservative.)

6. Compare QI from step 5 above with estimated intakes from breathing zone air samplers to be sure that estimates are of the proper magnitude. Study long-term correlation of air sample estimates of intake versus intakes QI estimated from urinalysis data by above procedure. Use this information to also ensure that individual air samplers are appropriately placed.

B.1.2 Intakes From *In-Vivo* Data by Simple Computation for Long-term (Chronic) Exposures Compared to Biological Half-life in Lung

These estimates will assume equilibrium between the translocation rate of Class Y material from the pulmonary spaces and the deposition rate in these spaces, as in the ICRP 2 lung model (ICRP 1960). This procedure will be conservative (result in a higher rate of intake) relative to the lung model used for ICRP 30 (1979), since the rate of translocation of the 12.5% of intake remaining in the deep lung occurs with a more rapid biological half-life of 120 d in the ICRP 2 model, compared to 15% remaining in the deep lung and translocated with a 500-d half-life (with a small portion to pulmonary lymph nodes) in the ICRP 30 model. Thus, the ICRP 2 model will require multiplication of the deposition rate in the deep lung by a larger factor to get intake. Also, the rate constant assumed for the translocation rate of a measured quantity of radioactivity in the lung will be larger.

The estimation of intake from *in-vivo* data proceeds as follows:

1. Take average of all lung burden estimates for the quarter (perhaps only one) for the minimum program. This is Q_{AV} in Bq in the lung.
2. Calculate rate of solubilization of uranium in the deep lung to be:

$$R = 0.693 Q_{AV}/120,$$

in Bq solubilized per day, assuming an average 120-d half-life for material solubilized and translocated from the deep lung. (This 120-d half-life is also closer to that typical of "Special Class Y" material; see Section A.3.10.)

3. Using this simplified equilibrium model, appropriately conservative for this estimation procedure, assume R is one-half the rate of deposition of Class Y, 1 micron AMAD, uranium in the deep (pulmonary) lung. Then, multiply R by $1/0.125 = 8$ to get the daily intake rate,

$$\begin{aligned} DI &= 8 \times 0.693 Q_{AV}/120 \\ &= 0.046 Q_{AV}, \text{ in units of Bq d}^{-1}. \end{aligned}$$

[The fraction 0.125 is the fraction of inhaled 1 micron AMAD, Type Y material, assumed to deposit in deep lung, where it can remain to be solubilized and translocated to blood (ICRP 1960; Wagner 1994; Cool et al. 1979).]

4. Assuming 65 workdays per quarter, but increasing the effective time of intake by 7/5 to maintain a balance with continuous excretion, the quarterly intake rate becomes:

$$\begin{aligned} QI &= 65 \times 0.046 Q_{AV} \times 7/5 \\ &= 4 Q_{AV}, \text{ in Bq intake per quarter.} \end{aligned}$$

Note: The quarter's intake rate becomes simply four times the average lung burden, using the above simplifying but conservative assumptions for the vast majority of cases. This could be applicable to many of the special Class Y exposures.

5. Compare the quarter's intake rate QI for each worker with the estimate from his breathing zone air sample measurements and maintain a long-term running correlation chart of these data. The plant health physicist can examine these correlations to aid in discovering any changes in air sample placement or operational safety conditions that might be needed, as well as for confirming the validity of both air monitoring and *in-vivo* counting data for Class Y uranium.

(It is assumed for the normal operating conditions far above the ambient average, the short-term excretion pattern following any single intakes of even Class Y material will not appreciably affect the *in-vivo* measurements carried out monthly or quarterly (Lawrence 1984). Furthermore, if an *in-vivo* measurement is made soon after such a limited single intake, the perturbation will tend to cause an overestimate of quarterly intake by the above simplified calculation procedure.)

6. Compare quarterly intakes with the quarterly intake limit of 10 CFR Part 20 (U.S. NRC 1994 1988, 1991).

Thus, since the DAC for insoluble material in the lung is independent of enrichment in alpha activity units (Lawrence 1984), the total maximum permissible quarterly intake for Class Y material is now (see Table A.2):

$$\begin{aligned} \text{QMPI} &= 2E(-11) \mu\text{Ci mL}^{-1} \times 10^7 \text{ ml d}^{-1} \times 65 \text{ days work/quarter} \\ &= 0.013 \mu\text{Ci (480 Bq)}. \end{aligned}$$

(Note that in ICRP 2, the MPC (occupational) for insoluble class material is $10^{-10} \mu\text{Ci mL}^{-1}$ (Table A.2). In this case, QMPI would be $0.065 \mu\text{Ci}$. Then, for Q_{AV} at equilibrium = $QI/4$ from above, the quarterly intake limit for chronic (assumed almost constant, long-term) exposure is the same as the maintenance of a "maximum permissible lung burden (MPLB)," as sometimes denoted [although not explicitly presented as such in ICRP 2 (1960)], of:

$$\text{MPLB} = \text{QMPI}/4 = 0.065/4 = 0.016 \mu\text{Ci (592 Bq)}.$$

This agrees with the value of $0.016 \mu\text{Ci}$ that can be calculated directly from dosimetric considerations that limit the 1,000 g lung to 0.3 rem wk^{-1} (ICRP 1960). This effectively checks this simple computational method. Any differences in values obtained by other authors (Fisher 1984; Bernard 1958; Scott and West 1975; Boback 1975) can be attributed to the rounding of MPCs to one digit.

These simple computational methods for evaluating *in-vivo* data will be practical only when lung burdens exceed routinely attainable MDAs.

B.2 Intake and Dose Estimation From Tables and Graphs

Dose estimates compatible with the more recent ICRP 54 methods can be obtained from tables or graphs that show the percent of intake excreted in the urine for the various solubility classes of uranium with time after acute exposure, or after onset of chronic exposures. Two graphs from the report of the DOE's Expert Group on Internal Dosimetry are shown as Fig. B.1 and Fig. B.2. These can be used to estimate intake, which can be translated into dose by ICRP 30 constants if the solubility of the exposure material and the times of exposure are known. For example, assume that an individual who worked in an area handling UF₆

(a Class W compound) gave a urine sample after a quarter of work when there were no known incidents or accidents and the sample showed a level of 0.1 Bq L^{-1} . It would be assumed that his dose could be best estimated from the "Intake Estimator – Urine – Chronic Inhalation," Fig. B.2. This figure gives about 0.09 Bq L^{-1} in urine at 90 d, for 1 Bq d^{-1} intake. Thus, an excretion rate of 0.1 Bq L^{-1} at 90 d under chronic exposure conditions for the quarter would correspond to $(0.1 \text{ Bq L}^{-1}) / (0.09 \text{ Bq L}^{-1} \text{ per Bq d}^{-1}) = 1.1 \text{ Bq d}^{-1}$ intake. When multiplied by the 65 workdays in a quarter, this gives a total intake of (1.1×90) or 100 Bq in the quarter. This intake can be converted to dose using the factor published in ICRP 30 of $1.8 \times 10^{-6} \text{ Sv Bq}^{-1}$ to get a dose of $1.8 \times 10^{-4} \text{ Sv}$. The conversion factors for Class Y and Class D are $3.3 \times 10^{-5} \text{ Sv Bq}^{-1}$ and $6.6 \times 10^{-7} \text{ Sv Bq}^{-1}$ weighted committed dose equivalent, respectively. A more precise quick estimate (certainly not called for in this situation) could be obtained by using the urine excretion rates in Table B.1, from the Canadian guidelines (Health and Welfare Canada 1987). Acute and chronic exposures can be evaluated alternatively using the tables from Lessard et al. (1987), such as those in Table B.2.

As time progresses and more data are obtained on an individual, it will be possible to better judge retention and excretion factors when there is a significant intake so that more sophisticated determinations of dose can be made. There are a number of computer programs that develop doses for both chronic and acute exposure situations (LaBone 1991).

In-vivo results can be interpreted using the Lessard et al. (1987) or Canadian tables (Health and Welfare Canada 1987), which show the levels at various times in the lung, assuming ICRP 30 parameters. However, this method is not sensitive enough to be useful except in cases where there is a very significant acute intake or where there is or has been relatively high long-term exposure to Class W or Y materials. As in the case of urinalyses, more sophisticated approaches can be used when there is additional data. In substantial acute exposure situations, urine, fecal, and *in-vivo* monitoring results must all be used together to best estimate the exposure.

Translocation rates and half-lives in the lung have been found to vary widely in actual inhalation cases involving non-transportable uranium (Scott and West 1975; Boback 1975; West et al. 1979; NCRP 1980; Lough 1958; Heatherton and Huesing 1958).

B.3 Special Bioassays Using Specific Human Data

Special bioassays to evaluate intakes and internal doses to individuals who are significantly exposed may sometimes be interpreted more accurately by using actual human data from published reports on accidental exposures. Such data are available for several important uranium compounds, and are presented in terms of specific retention and excretion functions that may be more applicable to the accurate evaluation of specific human exposures than the standard ICRP 30 models and functions.

Special bioassays to evaluate accurately the intakes and doses for individuals who have been exposed at levels approaching or exceeding regulatory limits, or requiring medical evaluation, are indicated by the action levels and actions of Tables 2 and 3. These special bioassays may often require careful confirmation by duplicate follow-up measurements at a specialized medical health physics evaluation center. Interpretation of the complete set of urine, fecal, and *in-vivo* counting data for each individual must be made by sequential fitting and testing of the models with all of the data, until a consistent metabolic pattern, and appropriate material balances, ensure that the best fit has been obtained to the applicable excretion-retention model. A "best estimate" of intake can then be calculated from the models (Lessard et al. 1987; Alexander et al. 1986; Lawrence 1984; Fisher 1984; Snyder 1958; McGuire 1983; LaTouche et al. 1987; Alexander 1974). Such modeling and intake estimation for the more significant exposure cases usually require the assistance and/or employment of a specialized medical health physics evaluation center, often utilizing some of the computerized models available (LaBone 1991).

However, certain simplified empirical equations can be used to model the excretion and *in-vivo* data for early estimates of intake, and the intake retention fraction (IRF) tables of Lessard et al. (1987) can also be used to estimate intakes for D, W, and/or Y classifications of material from quantities of material in lung, urine, or feces. The tables of Lessard et al. (1987) are based on a multi-compartment model of the body, the ICRP uptake retention fractions of material taken into blood that remain in organs of interest versus time, and coupling these uptake retention fractions with the ICRP lung model to estimate intake retention fractions (IRFs) according to the methods of Skrable et al. (1980). IRFs are tabulated for both cumulative and incremental daily quantities in urine and feces, as well as for lung contents. These IRFs have been checked against other sophisticated metabolic models by the distinguished set of co-authors of Lessard et al. (1987), who are leaders in model development for internal dose estimation. Also, examples of the use of these IRF tables and models are presented in the report. An example of the tabulation of IRF values for uranium is presented in Table B.2, abstracted from Lessard et al. (1987).

For early interpretation and reporting of special bioassay results, the IRFs of Table B.2 may be used to estimate intakes using early information about the Class (or proportion of Classes) of the material inhaled. When mixtures are involved, various proportions of Classes of D, W, and Y material will be assumed, with the use of Table B.2, until consistent estimates of intake are obtained. These intake estimates will then provide the basis for early reports to NRC or other regulatory agencies.

There are some other empirical formulas that may be used for early intake evaluation after an acute intake. For uranium hexafluoride and hydrolysis products, ICRP Publication 10 (1968) provided retention functions and action levels that have been used with single intakes. These retention functions can also be applied to any form of uranium for portions that have already entered blood as uranyl ion (Morrow et al. 1980). ICRP 10A (1971) provided retention functions and action levels for continuous intakes. [Both ICRP 10 reports have recently been superseded by ICRP 54 (1988).] Also, Fisher et al. (1990) have found that a "modified Wrenn Model" best fits the urinary excretion data for over 30 persons exposed to uranium hexafluoride and hydrolysis products.) The ICRP 10 model postulated that 80% of the soluble uranium taken up by the body initially is excreted the first day, with the remainder described by

$$Y_u = 0.1 t^{-1.5}$$

where Y_u is the fraction of the initial systemic uptake excreted on day t , for t greater than 1 d. Fisher et al. have found that a power function of the form $Y_u = 0.046 t^{-1.22}$ provides an excellent fit to the urinary excretion data in the UF₆ Sequoyah accident (Fisher et al. 1991).

Eckerman and Leggett (1986), in evaluating the Sequoyah uranium hexafluoride accident exposure, found that the ICRP 10 formulation overestimated somewhat the excretion at early times and cautioned that the ICRP 30 models are not suitable for relating excretion rates at early times to intake. They constructed an excretion function from blood to urine for uranyl compounds based on the human data of Bernard and Struxness (1957) combined with that of Hursh and Spoor (1973), and a specialized lung retention function for uranium hexafluoride exposure that, combined with their excretion function, seemed to fit most of the excretion patterns from the Sequoyah incident fairly well. It is cautioned, however, that this urinary excretion function is not designed to account for the bimodal retention that is sometimes observed in persons subject to lung edema (Moore and Kathren 1985; Kathren and Moore 1986). The urinary excretion function for blood is:

$$Y_u(t) = 7.3 \exp(-0.2t) + 2.4 \exp(-0.075t) + 0.08 \exp(-0.011t) \\ + 0.004 \exp(-0.0014t),$$

where $Y_u(t)$ is the percent per hour of an uptake into blood that is excreted in urine at time t in hours, from 30 min to several days after injection. The retention function for lung used in the Sequoyah analyses by Eckerman and Leggett (1986) was:

$$y(t) = 30 \exp(-0.01t/24) + 20 \exp(-0.5t/24),$$

where $y(t)$ is the percent of the initial single inhalation that was deposited in lung that remains in lung at t hours, the other portion having been transported to blood. An appropriate convolution integral of $Y(T-t)$ $(dy/dt)dt$ would then give the fraction deposited in the lung that is excreted per unit time at time T , which when integrated can be related to the intake (or deposition) retention fraction in any incremental urine sample. Caution should be taken to multiply the coefficients in the above two equations by 0.01 to convert them from percentages to fractions. The above excretion and retention functions, when appropriately fitted to the data, gave intakes in the Sequoyah incident that were usually well within a factor of two of those of Lessard and Fisher (Fisher 1986). The newer models of Fisher et al. (1990, 1991) appear to yield even better agreement for UF_6 exposures. Wrenn and Bertelli (1994) have proposed a somewhat different model. However, variations in retention functions for uranium involved in different physical or chemical processes must be taken into account for the most accurate estimation of intakes in the more significant cases (Thompson 1994; Raabe 1994; Forrest and Barber 1993; Fisher and Briant 1994; Wrenn and Bertelli 1994). Bernard (1977) has also developed a stochastic biokinetic model to describe the variability of excretion rates about average values.

Intake estimates may be obtained from multiple measurements over time using weighted least-squares methods presented in Lessard et al. (1987). Additional methods of estimating intakes are presented in Raabe (1994).

B.4 Internal Dose and Dose Commitment Estimates

When estimates of internal dose or dose commitment are required, they can be calculated from intake estimates for chronic low-exposure cases using the estimation methods discussed in previous sections, or in ICRP 30.

After all follow-up data are obtained from special or emergency bioassay cases, the intakes can be multiplied by the ICRP 30 dose-conversion factors, adjusted for differences between rate constants used by the ICRP and actual organ turnover rates and biological half-lives applicable to each individual. Individuals have often varied by factors of five or more from some of the major metabolic rates assumed in ICRP models. Thus, the long-term dose assessments in important cases must be made, taking into account the total metabolic pattern over time.

Table B.1 – Canadian Guide: Intake-Excretion Conversions. Retention (Bq) and Excretion (Bq d⁻¹) During Chronic Exposure

Class D

Time	Lung	Lymph	Gut	S Int	UL Int	LL Int	Bone Vol	Kidneys	Other	T Lung	T Body	Urinary	Fecal
0.0	0	0	0	0	0	0	0	0	0	0	0	0	0
.1	2.48E-02	0.16E-04	6.61E-04	5.32E-04	8.83E-05	3.46E-06	5.10E-04	2.75E-04	2.75E-04	2.51E-02	6.18E-02	2.74E-02	7.46E-06
.2	4.50E-02	1.15E-03	1.56E-03	2.35E-03	8.35E-04	6.94E-05	2.24E-03	1.20E-03	1.20E-03	4.65E-02	1.21E-01	5.55E-02	6.92E-05
.5	9.22E-02	5.53E-03	3.55E-03	9.82E-03	9.44E-03	2.20E-03	1.33E-02	7.07E-03	7.07E-03	9.77E-02	2.82E-01	1.21E-01	2.20E-01
1.0	1.37E-01	1.46E-02	5.22E-03	1.81E-02	3.73E-02	1.72E-02	4.48E-02	2.35E-02	2.35E-02	1.52E-01	5.14E-01	1.89E-01	1.72E-02
2.0	1.71E-01	2.76E-02	6.21E-03	2.33E-02	6.70E-02	6.95E-02	1.31E-01	6.66E-02	6.66E-02	1.99E-01	8.74E-01	2.55E-01	6.95E-02
5.0	1.82E-01	3.58E-02	6.42E-03	2.44E-02	8.11E-02	1.40E-01	4.21E-01	1.92E-01	1.92E-01	2.18E-01	1.54E+00	3.16E-01	1.45E-01
7.0	1.83E-01	3.60E-02	6.42E-03	2.44E-02	8.11E-02	1.40E-01	4.21E-01	2.57E-01	2.57E-01	2.19E-01	1.86E+00	3.37E-01	1.45E-01
14.0	1.83E-01	3.61E-02	6.42E-03	2.44E-02	8.11E-02	1.46E-01	1.17E+00	3.96E-01	3.96E-01	2.19E-01	2.70E+00	3.85E-01	1.46E-01
30.0	1.83E-01	3.61E-02	6.42E-03	2.44E-02	8.11E-02	1.46E-01	2.10E+00	4.92E-01	4.92E-01	2.19E-01	3.82E+00	4.33E-01	1.46E-01
60.0	1.83E-01	3.61E-02	6.42E-03	2.44E-02	8.11E-02	1.46E-01	3.62E+00	5.24E-01	5.24E-01	2.19E-01	5.44E+00	4.68E-01	1.46E-01
90.0	1.83E-01	3.61E-02	6.42E-03	2.44E-02	8.11E-02	1.46E-01	4.75E+00	5.46E-01	5.46E-01	2.19E-01	6.59E+00	4.72E-01	1.46E-01
180.0	1.83E-01	3.61E-02	6.42E-03	2.44E-02	8.11E-02	1.46E-01	6.74E+00	5.87E-01	5.87E-01	2.19E-01	8.66E+00	4.73E-01	1.46E-01
365.0	1.83E-01	3.61E-02	6.42E-03	2.44E-02	8.11E-02	1.46E-01	1.05E+01	6.58E-01	6.58E-01	2.19E-01	1.26E+01	4.73E-01	1.46E-01
730.0	1.83E-01	3.61E-02	6.42E-03	2.44E-02	8.11E-02	1.46E-01	2.07E+01	8.12E-01	8.12E-01	2.19E-01	2.31E+01	4.73E-01	1.46E-01
1825.0	1.83E-01	3.61E-02	6.42E-03	2.44E-02	8.11E-02	1.46E-01	3.46E+01	9.46E-01	9.46E-01	2.19E-01	3.72E+01	4.77E-01	1.46E-01
3650.0	1.83E-01	3.61E-02	6.42E-03	2.44E-02	8.11E-02	1.46E-01	5.38E+01	1.03E+00	1.03E+00	2.19E-01	5.66E+01	4.80E-01	1.46E-01
7300.0	1.83E-01	3.61E-02	6.42E-03	2.44E-02	8.11E-02	1.46E-01	7.64E+01	1.05E+00	1.05E+00	2.19E-01	7.92E+01	4.83E-01	1.46E-01
18250.0	1.83E-01	3.61E-02	6.42E-03	2.44E-02	8.11E-02	1.46E-01							

Class W

Time	Lung	Lymph	Gut	S Int	UL Int	LL Int	Bone Vol	Kidneys	Other	T Lung	T Body	Urinary	Fecal
0.0	0	0	0	0	0	0	0	0	0	0	0	0	0
.1	2.89E-02	8.66E-07	1.47E-03	1.18E-03	1.96E-04	7.71E-06	1.50E-04	8.10E-05	8.10E-05	2.89E-02	6.26E-02	7.94E-03	7.71E-06
.2	5.61E-02	3.46E-06	3.49E-03	5.25E-03	1.86E-03	1.55E-04	6.37E-04	3.42E-04	3.42E-04	5.61E-02	1.24E-01	1.53E-02	1.55E-04
.5	1.32E-01	2.16E-05	8.08E-03	2.22E-02	2.12E-02	4.93E-03	3.49E-03	1.86E-03	1.86E-03	1.32E-01	3.06E-01	2.98E-02	4.93E-03
1.0	2.44E-01	8.58E-05	1.23E-02	4.22E-02	7.74E-02	3.91E-02	1.02E-02	5.64E-03	5.64E-03	2.44E-01	5.93E-01	4.18E-02	3.91E-02
2.0	4.37E-01	3.40E-04	1.55E-02	5.75E-02	1.62E-01	1.64E-01	2.86E-02	1.44E-02	1.44E-02	4.37E-01	1.08E+00	5.10E-02	1.64E-01
5.0	9.06E-01	2.07E-03	1.72E-02	6.51E-02	2.15E-01	3.65E-01	8.42E-02	3.81E-02	3.81E-02	9.06E-01	1.92E+00	6.11E-02	3.65E-01
7.0	1.19E+00	3.98E-03	1.74E-02	6.61E-02	2.20E-01	3.89E-01	1.20E-01	5.04E-02	5.04E-02	1.19E+00	2.27E+00	6.57E-02	3.89E-01
14.0	2.10E+00	1.49E-02	1.78E-02	6.76E-02	2.25E-01	4.04E-01	2.32E-01	7.85E-02	7.85E-02	2.10E+00	3.41E+00	7.75E-02	4.04E-01
30.0	3.88E+00	5.94E-02	1.85E-02	7.02E-02	2.34E-01	4.20E-01	4.34E-01	1.04E-01	1.04E-01	3.94E+00	5.51E+00	9.30E-02	4.20E-01
60.0	6.31E+00	1.83E-01	1.94E-02	7.38E-02	2.46E-01	4.42E-01	6.93E-01	1.21E-01	1.21E-01	6.49E+00	8.41E+00	1.09E-01	4.42E-01
90.0	7.92E+00	3.20E-01	2.00E-02	7.62E-02	2.54E-01	4.56E-01	8.75E-01	1.33E-01	1.33E-01	8.24E+00	1.04E+01	1.19E-01	4.56E-01
180.0	1.01E+01	6.42E-01	2.09E-02	7.94E-02	2.65E-01	4.76E-01	1.27E+00	1.54E-01	1.54E-01	1.08E+01	1.34E+01	1.34E-01	4.76E-01
365.0	1.10E+01	8.67E-01	2.12E-02	8.07E-02	2.69E-01	4.84E-01	1.92E+00	1.73E-01	1.73E-01	1.18E+01	1.52E+01	1.41E-01	4.84E-01
730.0	1.10E+01	9.01E-01	2.12E-02	8.07E-02	2.69E-01	4.84E-01	3.06E+00	1.96E-01	1.96E-01	1.19E+01	1.64E+01	1.42E-01	4.84E-01
1825.0	1.10E+01	9.02E-01	2.12E-02	8.07E-02	2.69E-01	4.84E-01	6.44E+00	2.43E-01	2.43E-01	1.19E+01	1.96E+01	1.43E-01	4.84E-01
3650.0	1.10E+01	9.02E-01	2.12E-02	8.07E-02	2.69E-01	4.84E-01	1.03E+01	2.84E-01	2.84E-01	1.19E+01	2.39E+01	1.43E-01	4.84E-01
7300.0	1.10E+01	9.02E-01	2.12E-02	8.07E-02	2.69E-01	4.84E-01	1.61E+01	3.09E-01	3.09E-01	1.19E+01	2.97E+01	1.44E-01	4.84E-01
18250.0	1.10E+01	9.02E-01	2.12E-02	8.07E-02	2.69E-01	4.84E-01	2.30E+01	3.15E-01	3.15E-01	1.19E+01	3.66E+01	1.45E-01	4.84E-01

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Table B.1 continued - Retention (Bq) and Excretion (Bq d⁻¹) During Chronic Exposure

Class Y	Lung	Lymph	Gut	S Int	UL Int	LL Int	Bone Vol	Kidneys	Other	T Lung	T Body	Urinary	Fecal
0.0	0	0	0	0	0	0	0	0	0	0	0	0	0
.1	3.17E-02	2.60E-07	1.89E-03	1.53E-03	2.55E-04	1.02E-05	8.15E-06	4.39E-06	4.39E-06	3.17E-02	6.30E-02	4.31E-04	1.02E-05
.2	6.12E-02	1.04E-06	4.44E-03	6.82E-03	2.41E-03	2.01E-04	3.45E-05	1.86E-05	1.86E-05	6.12E-02	1.26E-01	8.30E-04	2.01E-04
.5	1.41E-01	6.50E-06	9.98E-03	2.86E-02	2.74E-02	6.38E-03	1.89E-04	1.01E-04	1.01E-04	1.41E-01	3.14E-01	1.61E-03	6.38E-03
1.0	2.55E-01	2.60E-05	1.47E-02	5.32E-02	9.85E-02	5.01E-02	5.80E-04	3.04E-04	3.04E-04	2.55E-01	6.16E-01	2.23E-03	5.01E-02
2.0	4.51E-01	1.04E-04	1.81E-02	7.08E-02	2.01E-01	2.06E-01	1.52E-03	7.70E-04	7.70E-04	4.51E-01	1.12E+00	2.69E-03	2.06E-01
5.0	9.38E-01	6.48E-04	1.97E-02	7.85E-02	2.60E-01	4.43E-01	4.43E-03	2.00E-03	2.00E-03	9.39E-01	1.92E+00	3.18E-03	4.43E-01
7.0	1.24E+00	1.27E-03	1.98E-02	7.92E-02	2.63E-01	4.69E-01	6.26E-03	2.63E-03	2.63E-03	1.24E+00	2.26E+00	3.41E-03	4.69E-01
14.0	2.28E+00	5.05E-03	1.99E-02	7.95E-02	2.65E-01	4.77E-01	1.20E-02	4.04E-03	4.04E-03	2.28E+00	3.32E+00	3.97E-03	4.77E-01
30.0	4.60E+00	2.29E-02	2.02E-02	8.05E-02	2.68E-01	4.83E-01	2.21E-02	5.24E-03	5.24E-03	4.60E+00	5.68E+00	4.67E-03	4.79E-01
60.0	8.83E+00	8.99E-02	2.03E-02	8.11E-02	2.70E-01	4.87E-01	4.39E-02	6.04E-03	6.04E-03	8.83E+00	1.00E+01	5.43E-03	4.83E-01
90.0	1.29E+01	1.98E-01	2.08E-02	8.29E-02	2.76E-01	4.97E-01	6.91E-02	8.73E-03	8.73E-03	1.29E+01	1.42E+01	6.03E-03	4.87E-01
180.0	2.41E+01	7.47E-01	2.08E-02	8.29E-02	2.76E-01	4.97E-01	6.91E-02	8.73E-03	8.73E-03	2.48E+01	2.60E+01	7.68E-03	4.97E-01
365.0	4.32E+01	2.73E+00	2.15E-02	8.58E-02	2.86E-01	5.15E-01	1.29E-01	1.32E-02	1.32E-02	4.59E+01	4.71E+01	1.11E-02	5.15E-01
730.0	6.91E+01	8.69E+00	2.25E-02	8.98E-02	2.99E-01	5.39E-01	2.87E-01	2.25E-02	2.25E-02	7.78E+01	7.92E+01	1.76E-02	5.39E-01
1825.0	9.98E+01	2.94E+01	2.37E-02	9.45E-02	3.15E-01	5.67E-01	9.78E-01	4.95E-02	4.95E-02	1.29E+02	1.31E+02	3.30E-02	5.67E-01
3650.0	1.08E+02	5.22E+01	2.40E-02	9.57E-02	3.19E-01	5.74E-01	2.40E+00	8.07E-02	8.07E-02	1.60E+02	1.64E+02	4.51E-02	5.43E-01
7300.0	1.08E+02	7.26E+01	2.40E-02	9.58E-02	3.19E-01	5.75E-01	4.83E+00	1.04E-01	1.04E-01	1.81E+02	1.87E+02	5.02E-02	5.75E-01
18250.0	1.08E+02	1.14E+02	2.40E-02	9.58E-02	3.19E-01	5.75E-01	7.88E+00	1.11E-01	1.11E-01	2.22E+02	2.32E+02	5.10E-02	5.75E-01

Table B.2 – Intake Retention Fractions for Body and Excretion Compartments
(Lessard et al. 1987). Tables for ^{235}U are the same.

Class D Time After Single Intake (Days)	AMAD = 1 Micron		Half-life = 1.16E+12 Days			^{238}U
	Fraction of Initial Intake In:					
	Systemic Organs	Lungs	Nasal Passages	GI Tract	Total Body	
1.00E-01	2.35E-01	2.26E-01	3.03E-04	1.54E-01	6.15E-01	
2.00E-01	2.35E-01	2.01E-01	2.96E-07	1.52E-01	5.88E-01	
3.00E-01	2.35E-01	1.79E-01	0.00E+00	1.49E-01	5.63E-01	
4.00E-01	2.33E-01	1.60E-01	0.00E+00	1.45E-01	5.38E-01	
5.00E-01	2.31E-01	1.43E-01	0.00E+00	1.41E-01	5.15E-01	
6.00E-01	2.30E-01	1.27E-01	0.00E+00	1.35E-01	4.92E-01	
7.00E-01	2.28E-01	1.13E-01	0.00E+00	1.29E-01	4.70E-01	
8.00E-01	2.26E-01	1.01E-01	0.00E+00	1.22E-01	4.49E-01	
9.00E-01	2.24E-01	8.96E-02	0.00E+00	1.16E-01	4.29E-01	
1.00E+00	2.22E-01	7.97E-02	0.00E+00	1.09E-01	4.10E-01	
2.00E+00	2.05E-01	2.42E-02	0.00E+00	4.95E-02	2.78E-01	
3.00E+00	1.90E-01	7.13E-03	0.00E+00	1.97E-02	2.16E-01	
4.00E+00	1.76E-01	2.05E-03	0.00E+00	7.49E-03	1.86E-01	
5.00E+00	1.65E-01	5.81E-04	0.00E+00	2.79E-03	1.68E-01	
6.00E+00	1.54E-01	1.62E-04	0.00E+00	1.03E-03	1.55E-01	
7.00E+00	1.45E-01	4.47E-05	0.00E+00	3.81E-04	1.45E-01	
8.00E+00	1.36E-01	1.22E-05	0.00E+00	1.40E-04	1.36E-01	
9.00E+00	1.28E-01	3.32E-06	0.00E+00	5.16E-05	1.28E-01	
1.00E+01	1.21E-01	8.96E-07	0.00E+00	1.90E-05	1.21E-01	
2.00E+01	7.36E-02	0.00E+00	0.00E+00	0.00E+00	7.36E-02	
3.00E+01	5.06E-02	0.00E+00	0.00E+00	0.00E+00	5.06E-02	
4.00E+01	3.77E-02	0.00E+00	0.00E+00	0.00E+00	3.77E-02	
5.00E+01	2.96E-02	0.00E+00	0.00E+00	0.00E+00	2.96E-02	
6.00E+01	2.42E-02	0.00E+00	0.00E+00	0.00E+00	2.42E-02	
7.00E+01	2.04E-02	0.00E+00	0.00E+00	0.00E+00	2.04E-02	
8.00E+01	1.78E-02	0.00E+00	0.00E+00	0.00E+00	1.78E-02	
9.00E+01	1.60E-02	0.00E+00	0.00E+00	0.00E+00	1.60E-02	
1.00E+02	1.47E-02	0.00E+00	0.00E+00	0.00E+00	1.47E-02	
2.00E+02	1.15E-02	0.00E+00	0.00E+00	0.00E+00	1.15E-02	
3.00E+02	1.12E-02	0.00E+00	0.00E+00	0.00E+00	1.12E-02	
4.00E+02	1.10E-02	0.00E+00	0.00E+00	0.00E+00	1.10E-02	
5.00E+02	1.09E-02	0.00E+00	0.00E+00	0.00E+00	1.09E-02	
6.00E+02	1.07E-02	0.00E+00	0.00E+00	0.00E+00	1.07E-02	
7.00E+02	1.06E-02	0.00E+00	0.00E+00	0.00E+00	1.06E-02	
8.00E+02	1.04E-02	0.00E+00	0.00E+00	0.00E+00	1.04E-02	
9.00E+02	1.02E-02	0.00E+00	0.00E+00	0.00E+00	1.02E-02	
1.00E+03	1.01E-02	0.00E+00	0.00E+00	0.00E+00	1.01E-02	
2.00E+03	8.71E-03	0.00E+00	0.00E+00	0.00E+00	8.71E-03	
4.00E+03	6.53E-03	0.00E+00	0.00E+00	0.00E+00	6.53E-03	
6.00E+03	4.92E-03	0.00E+00	0.00E+00	0.00E+00	4.92E-03	
8.00E+03	3.72E-03	0.00E+00	0.00E+00	0.00E+00	3.72E-03	
1.00E+04	2.81E-03	0.00E+00	0.00E+00	0.00E+00	2.81E-03	
2.00E+04	7.02E-04	0.00E+00	0.00E+00	0.00E+00	7.02E-04	

Table B.2 (continued)

Class D	AMAD = 1 Micron	Half-life = 1.16E+12 Days		²³⁸ U
Time After Single Intake (Days)	Fraction of Initial Intake In:			
	24-Hour Urine	Accumulated Urine	24-Hour Feces	Accumulated Feces
1.00E-01		2.41E-02		6.04E-05
2.00E-01		5.00E-02		7.338E-04
3.00E-01		7.36E-02		2.60E-03
4.00E-01		9.50E-02		5.81E-03
5.00E-01		1.14E-01		1.02E-02
6.00E-01		1.32E-01		1.56E-02
7.00E-01		1.48E-01		2.18E-02
8.00E-01		1.62E-01		2.84E-02
9.00E-01		1.75E-01		3.54E-02
1.00E+00	1.87E-01	1.87E-01	4.24E-02	4.24E-02
2.00E+00	7.27E-02	2.59E-01	5.92E-02	1.02E-01
3.00E+00	3.21E-02	2.91E-01	2.98E-02	1.31E-01
4.00E+00	1.82E-02	3.10E-01	1.22E-02	1.44E-01
5.00E+00	1.31E-02	3.23E-01	4.70E-03	1.48E-01
6.00E+00	1.09E-02	3.34E-01	1.76E-03	1.50E-01
7.00E+00	9.64E-03	3.43E-01	6.52E-04	1.51E-01
8.00E+00	8.71E-03	3.52E-01	2.41E-04	1.51E-01
9.00E+00	7.94E-03	3.60E-01	8.87E-05	1.51E-01
1.00E+01	7.26E-03	3.67E-01	3.26E-05	1.51E-01
2.00E+01	3.26E-03	4.14E-01	0.00E+00	1.51E-01
3.00E+01	1.71E-03	4.37E-01	0.00E+00	1.51E-01
4.00E+01	1.03E-03	4.50E-01	0.00E+00	1.51E-01
5.00E+01	6.67E-04	4.59E-01	0.00E+00	1.51E-01
6.00E+01	4.54E-04	4.64E-01	0.00E+00	1.51E-01
7.00E+01	3.16E-04	4.68E-01	0.00E+00	1.51E-01
8.00E+01	2.22E-04	4.70E-01	0.00E+00	1.51E-01
9.00E+01	1.57E-04	4.72E-01	0.00E+00	1.51E-01
1.00E+02	1.11E-04	4.73E-01	0.00E+00	1.51E-01
2.00E+02	5.15E-06	4.77E-01	0.00E+00	1.51E-01
3.00E+02	1.80E-06	4.77E-01	0.00E+00	1.51E-01
4.00E+02	1.67E-06	4.77E-01	0.00E+00	1.51E-01
5.00E+02	1.64E-06	4.77E-01	0.00E+00	1.51E-01
6.00E+02	1.61E-06	4.77E-01	0.00E+00	1.51E-01
7.00E+02	1.58E-06	4.78E-01	0.00E+00	1.51E-01
8.00E+02	1.56E-06	4.78E-01	0.00E+00	1.51E-01
9.00E+02	1.53E-06	4.78E-01	0.00E+00	1.51E-01
1.00E+03	1.50E-06	4.78E-01	0.00E+00	1.51E-01
2.00E+03	1.27E-06	4.79E-01	0.00E+00	1.51E-01
4.00E+03	9.35E-07	4.82E-01	0.00E+00	1.51E-01
8.00E+03	5.27E-07	4.84E-01	0.00E+00	1.51E-01
1.00E+04	4.01E-07	4.85E-01	0.00E+00	1.51E-01
2.00E+04	1.17E-07	4.87E-01	0.00E+00	1.51E-01

Table B.2 (continued)

Class W	AMAD = 1 Micron		Half-life = 1.16E+12 Days		²³⁸ U
Time After Single Intake (Days)	Fraction of Initial Intake In:				
	Systemic Organs	Lungs	Nasal Passages	GI Tract	Total Body
1.00E-01	6.44E-02	2.76E-01	2.35E-01	5.68E-02	6.32E-01
2.00E-01	5.91E-02	2.65E-01	1.97E-01	1.04E-01	6.25E-01
3.00E-01	5.54E-02	2.56E-01	1.66E-01	1.42E-01	6.19E-01
4.00E-01	5.26E-02	2.47E-01	1.39E-01	1.73E-01	6.13E-01
5.00E-01	5.04E-02	2.40E-01	1.17E-01	1.99E-01	6.06E-01
6.00E-01	4.87E-02	2.33E-01	9.86E-02	2.18E-01	5.98E-01
7.00E-01	4.72E-02	2.27E-01	8.29E-02	2.33E-01	5.90E-01
8.00E-01	4.60E-02	2.21E-01	6.97E-02	2.43E-01	5.80E-01
9.00E-01	4.49E-02	2.16E-01	5.86E-02	2.50E-01	5.69E-01
1.00E+00	4.39E-02	2.11E-01	4.93E-02	2.53E-01	5.57E-01
2.00E+00	3.85E-02	1.77E-01	8.72E-03	1.93E-01	4.17E-01
3.00E+00	3.58E-02	1.60E-01	1.54E-03	1.07E-01	3.04E-01
4.00E+00	3.36E-02	1.50E-01	2.72E-04	5.40E-02	2.38E-01
5.00E+00	3.18E-02	1.45E-01	4.82E-05	2.69E-02	2.03E-01
6.00E+00	3.00E-02	1.41E-01	8.51E-06	1.39E-02	1.85E-01
7.00E+00	2.85E-02	1.38E-01	1.51E-06	7.70E-03	1.74E-01
8.00E+00	2.70E-02	1.36E-01	2.66E-07	4.77E-03	1.68E-01
9.00E+00	2.57E-02	1.34E-01	4.70E-08	3.36E-03	1.63E-01
1.00E+01	2.45E-02	1.32E-01	0.00E+00	2.67E-03	1.59E-01
2.00E+01	1.66E-02	1.16E-01	0.00E+00	1.78E-03	1.35E-01
3.00E+01	1.27E-02	1.02E-01	0.00E+00	1.55E-03	1.17E-01
4.00E+01	1.05E-02	9.01E-02	0.00E+00	1.35E-03	1.02E-01
5.00E+01	9.03E-03	7.93E-02	0.00E+00	1.17E-03	8.95E-02
6.00E+01	7.98E-03	6.98E-02	0.00E+00	1.02E-03	7.88E-02
7.00E+01	7.19E-03	6.14E-02	0.00E+00	8.90E-04	6.95E-02
8.00E+01	6.59E-03	5.40E-02	0.00E+00	7.75E-04	6.14E-02
9.00E+01	6.11E-03	4.75E-02	0.00E+00	6.75E-04	5.43E-02
1.00E+02	5.73E-03	4.18E-02	0.00E+00	5.87E-04	4.81E-02
2.00E+02	4.06E-03	1.15E-02	0.00E+00	1.47E-04	1.57E-02
3.00E+02	3.58E-03	3.15E-03	0.00E+00	3.67E-05	6.77E-03
4.00E+02	3.39E-03	8.55E-04	0.00E+00	9.18E-06	4.26E-03
5.00E+02	3.30E-03	2.31E-04	0.00E+00	2.29E-06	3.54E-03
6.00E+02	3.24E-03	6.19E-05	0.00E+00	5.74E-07	3.31E-03
7.00E+02	3.19E-03	1.65E-05	0.00E+00	1.43E-07	3.21E-03
8.00E+02	3.14E-03	4.40E-06	0.00E+00	3.58E-08	3.15E-03
9.00E+02	3.10E-03	1.16E-06	0.00E+00	0.00E+00	3.10E-03
1.00E+03	3.05E-03	3.08E-07	0.00E+00	0.00E+00	3.05E-03
3.00E+03	2.28E-03	0.00E+00	0.00E+00	0.00E+00	2.28E-03
5.00E+03	1.71E-03	0.00E+00	0.00E+00	0.00E+00	1.71E-03
7.00E+03	1.29E-03	0.00E+00	0.00E+00	0.00E+00	1.29E-03
9.00E+03	9.76E-04	0.00E+00	0.00E+00	0.00E+00	9.76E-04
1.00E+04	8.49E-04	0.00E+00	0.00E+00	0.00E+00	8.49E-04

Table B.2 (continued)

Class W Time After Single Intake (Days)	AMAD = 1 Micron		Half-life = 1.16E+12 Days		²³⁸ U
	Fraction Of Initial Intake In:				
	24-Hour Urine	Accumulated Urine	24-Hour Feces	Accumulated Feces	
1.00E-01		6.93E-03		8.06E-06	
2.00E-01		1.37E-02		1.62E-04	
3.00E-01		1.92E-02		8.17E-04	
4.00E-01		2.40E-02		2.38E-03	
5.00E-01		2.80E-02		5.16E-03	
6.00E-01		3.15E-02		9.38E-03	
7.00E-01		3.44E-02		1.51E-02	
8.00E-01		3.70E-02		2.23E-02	
9.00E-01		3.93E-02		3.09E-02	
1.00E+00	4.13E-02	4.13E-02	4.07E-02	4.07E-02	
2.00E+00	1.09E-02	5.21E-02	1.29E-01	1.70E-01	
3.00E+00	4.72E-03	5.69E-02	1.08E-01	2.78E-01	
4.00E+00	3.22E-03	6.01E-02	6.28E-02	3.41E-01	
5.00E+00	2.69E-03	6.28E-02	3.23E-02	3.73E-01	
6.00E+00	2.40E-03	6.52E-02	1.61E-02	3.89E-01	
7.00E+00	2.19E-03	6.74E-02	8.29E-03	3.98E-01	
8.00E+00	2.02E-03	6.94E-02	4.57E-03	4.02E-01	
9.00E+00	1.88E-03	7.13E-02	2.81E-03	4.05E-01	
1.00E+01	1.75E-03	7.30E-02	1.96E-03	4.07E-01	
2.00E+01	1.03E-03	8.60E-02	1.03E-03	4.19E-01	
3.00E+01	7.28E-04	9.44E-02	8.97E-04	4.28E-01	
4.00E+01	5.75E-04	1.01E-01	7.81E-04	4.36E-01	
5.00E+01	4.80E-04	1.06E-01	6.80E-04	4.44E-01	
6.00E+01	4.11E-04	1.10E-01	5.92E-04	4.50E-01	
7.00E+01	3.57E-04	1.14E-01	5.15E-04	4.55E-01	
8.00E+01	3.12E-04	1.18E-01	4.48E-04	4.60E-01	
9.00E+01	2.75E-04	1.20E-01	3.90E-04	4.64E-01	
1.00E+02	2.43E-04	1.23E-01	3.40E-04	4.68E-01	
2.00E+02	7.49E-05	1.37E-01	8.50E-05	4.86E-01	
3.00E+02	2.33E-05	1.41E-01	2.12E-05	4.91E-01	
4.00E+02	7.28E-06	1.43E-01	5.31E-06	4.92E-01	
5.00E+02	2.46E-06	1.43E-01	1.33E-06	4.92E-01	
6.00E+02	1.05E-06	1.43E-01	3.32E-07	4.92E-01	
7.00E+02	6.35E-07	1.44E-01	8.30E-08	4.92E-01	
8.00E+02	5.13E-07	1.44E-01	2.07E-08	4.92E-01	
9.00E+02	4.74E-07	1.44E-01	0.00E+00	4.92E-01	
1.00E+03	4.58E-07	1.44E-01	0.00E+00	4.92E-01	
3.00E+03	3.28E-07	1.44E-01	0.00E+00	4.92E-01	
5.00E+03	2.42E-07	1.45E-01	0.00E+00	4.92E-01	
7.00E+03	1.81E-07	1.45E-01	0.00E+00	4.92E-01	
9.00E+03	1.36E-07	1.46E-01	0.00E+00	4.92E-01	
1.00E+04	1.18E-07	1.46E-01	0.00E+00	4.92E-01	

Table B.2 (continued)

Class Y	AMAD =		Half-life = 1.16E+12 Days			²³⁸ U
	1 Micron		Fraction of Initial Intake In:			
Time After Single Intake (Days)	Systemic Organs	Lungs	Nasal Passages	GI Tract	Total Body	
1.00E-01	3.54E-03	3.04E-01	2.58E-01	7.30E-02	6.39E-01	
2.00E-01	3.24E-03	2.85E-01	2.17E-01	1.33E-01	6.38E-01	
3.00E-01	3.02E-03	2.70E-01	1.82E-01	1.82E-01	6.37E-01	
4.00E-01	2.86E-03	2.57E-01	1.53E-01	2.21E-01	6.35E-01	
5.00E-01	2.73E-03	2.47E-01	1.29E-01	2.52E-01	6.31E-01	
6.00E-01	2.63E-03	2.38E-01	1.08E-01	2.76E-01	6.25E-01	
7.00E-01	2.54E-03	2.31E-01	9.12E-02	2.93E-01	6.18E-01	
8.00E-01	2.46E-03	2.24E-01	7.67E-02	3.05E-01	6.09E-01	
9.00E-01	2.40E-03	2.19E-01	6.45E-02	3.12E-01	5.97E-01	
1.00E+00	2.34E-03	2.13E-01	5.42E-02	3.15E-01	5.85E-01	
2.00E+00	2.02E-03	1.80E-01	9.59E-03	2.32E-01	4.24E-01	
3.00E+00	1.87E-03	1.65E-01	1.70E-03	1.25E-01	2.93E-01	
4.00E+00	1.76E-03	1.57E-01	3.00E-04	6.06E-02	2.19E-01	
5.00E+00	1.65E-03	1.53E-01	5.30E-05	2.85E-02	1.83E-01	
6.00E+00	1.56E-03	1.51E-01	9.37E-06	1.34E-02	1.66E-01	
7.00E+00	1.47E-03	1.49E-01	1.66E-06	6.42E-03	1.57E-01	
8.00E+00	1.39E-03	1.49E-01	2.93E-07	3.17E-03	1.53E-01	
9.00E+00	1.32E-03	1.48E-01	5.17E-08	1.64E-03	1.51E-01	
1.00E+01	1.25E-03	1.48E-01	0.00E+00	9.19E-04	1.50E-01	
2.00E+01	8.23E-04	1.46E-01	0.00E+00	2.36E-04	1.48E-01	
3.00E+01	6.20E-04	1.45E-01	0.00E+00	2.32E-04	1.46E-01	
4.00E+01	5.08E-04	1.43E-01	0.00E+00	2.29E-04	1.44E-01	
5.00E+01	4.40E-04	1.42E-01	0.00E+00	2.26E-04	1.43E-01	
6.00E+01	3.97E-04	1.41E-01	0.00E+00	2.23E-04	1.41E-01	
7.00E+01	3.69E-04	1.39E-01	0.00E+00	2.19E-04	1.40E-01	
8.00E+01	3.50E-04	1.38E-01	0.00E+00	2.16E-04	1.38E-01	
9.00E+01	3.38E-04	1.36E-01	0.00E+00	2.13E-04	1.37E-01	
1.00E+02	3.32E-04	1.35E-01	0.00E+00	2.11E-04	1.35E-01	
2.00E+02	3.50E-04	1.22E-01	0.00E+00	1.83E-04	1.22E-01	
3.00E+02	3.92E-04	1.10E-01	0.00E+00	1.60E-04	1.11E-01	
4.00E+02	4.33E-04	9.98E-02	0.00E+00	1.39E-04	1.00E-01	
5.00E+02	4.71E-04	9.06E-02	0.00E+00	1.21E-04	9.12E-02	
6.00E+02	5.08E-04	8.23E-02	0.00E+00	1.05E-04	8.29E-02	
7.00E+02	5.42E-04	7.49E-02	0.00E+00	9.16E-05	7.55E-02	
8.00E+02	5.74E-04	6.83E-02	0.00E+00	7.98E-05	6.89E-02	
9.00E+02	6.03E-04	6.23E-02	0.00E+00	6.94E-05	6.30E-02	
1.00E+03	6.30E-04	5.70E-02	0.00E+00	6.05E-05	5.77E-02	
2.00E+03	7.97E-04	2.55E-02	0.00E+00	1.51E-05	2.63E-02	
3.00E+03	8.27E-04	1.34E-02	0.00E+00	3.78E-06	1.42E-02	
4.00E+03	7.87E-04	8.25E-03	0.00E+00	9.45E-07	9.04E-03	
5.00E+03	7.18E-04	5.92E-03	0.00E+00	2.36E-07	6.63E-03	
7.00E+03	5.66E-04	4.27E-03	0.00E+00	1.48E-08	4.83E-03	
8.00E+03	4.96E-04	4.00E-03	0.00E+00	0.00E+00	4.50E-03	
9.00E+03	4.33E-04	3.87E-03	0.00E+00	0.00E+00	4.30E-03	
1.00E+04	3.78E-04	3.80E-03	0.00E+00	0.00E+00	4.18E-03	
2.00E+04	9.44E-05	3.74E-03	0.00E+00	0.00E+00	3.83E-03	

Table B.2 (continued)

Class Y	AMAD = 1 Micron	Half-life = 1.16E+12 Days		²³⁸ U
Time After Single Intake (Days)	Fraction of Initial Intake In:			
	24-Hour Urine	Accumulated Urine	24-Hour Feces	Accumulated Feces
1.00E-01		3.80E-04		1.04E-05
2.00E-01		7.49E-04		2.10E-04
3.00E-01		1.05E-03		1.06E-03
4.00E-01		1.31E-03		3.07E-03
5.00E-01		1.53E-03		6.66E-03
6.00E-01		1.71E-03		1.21E-02
7.00E-01		1.87E-03		1.94E-02
8.00E-01		2.01E-03		2.86E-02
9.00E-01		2.13E-03		3.95E-02
1.00E+00	2.23E-03	2.23E-03	5.20E-02	5.20E-02
2.00E+00	5.49E-04	2.78E-03	1.60E-01	2.13E-01
3.00E+00	2.30E-04	3.01E-03	1.31E-01	3.43E-01
4.00E+00	1.57E-04	3.17E-03	7.35E-02	4.17E-01
5.00E+00	1.31E-04	3.30E-03	3.63E-02	4.53E-01
6.00E+00	1.17E-04	3.41E-03	1.72E-02	4.70E-01
7.00E+00	1.07E-04	3.52E-03	8.11E-03	4.78E-01
8.00E+00	9.81E-05	3.62E-03	3.88E-03	4.82E-01
9.00E+00	9.07E-05	3.71E-03	1.91E-03	4.84E-01
1.00E+01	8.42E-05	3.79E-03	9.83E-04	4.85E-01
2.00E+01	4.69E-05	4.40E-03	1.35E-04	4.87E-01
3.00E+01	3.27E-05	4.78E-03	1.33E-04	4.89E-01
4.00E+01	2.65E-05	5.07E-03	1.31E-04	4.90E-01
5.00E+01	2.34E-05	5.32E-03	1.29E-04	4.91E-01
7.00E+01	2.04E-05	5.75E-03	1.26E-04	4.94E-01
8.00E+01	1.96E-05	5.95E-03	1.24E-04	4.95E-01
9.00E+01	1.91E-05	6.14E-03	1.22E-04	4.96E-01
1.00E+02	1.87E-05	6.33E-03	1.20E-04	4.97E-01
2.00E+02	1.81E-05	8.15E-03	1.05E-04	5.09E-01
3.00E+02	1.83E-05	9.97E-03	9.13E-05	5.18E-01
4.00E+02	1.82E-05	1.18E-02	7.95E-05	5.27E-01
5.00E+02	1.81E-05	1.36E-02	6.92E-05	5.34E-01
7.00E+02	1.74E-05	1.72E-02	5.24E-05	5.46E-01
8.00E+02	1.69E-05	1.89E-02	4.56E-05	5.51E-01
9.00E+02	1.64E-05	2.05E-02	3.97E-05	5.56E-01
1.00E+03	1.58E-05	2.21E-02	3.46E-05	5.59E-01
2.00E+03	9.76E-06	3.49E-02	8.65E-06	5.78E-01
3.00E+03	5.39E-06	4.23E-02	2.16E-06	5.83E-01
4.00E+03	2.86E-06	4.63E-02	5.40E-07	5.84E-01
5.00E+03	1.50E-06	4.84E-02	1.35E-07	5.84E-01
7.00E+03	4.36E-07	5.01E-02	0.00E+00	5.84E-01
8.00E+03	2.48E-07	5.04E-02	0.00E+00	5.84E-01
9.00E+03	1.50E-07	5.06E-02	0.00E+00	5.84E-01
1.00E+04	9.75E-08	5.07E-02	0.00E+00	5.84E-01
2.00E+04	1.31E-08	5.11E-02	0.00E+00	5.84E-01

Table B.2 (continued)

Ingestion	$f_1=5.00E-02$	Half-life = 1.16E+12 Days	²³⁸ U
Time After Single Intake (Days)	Fraction of Initial Intake In:		
	System Organs	GI Tract	Total Body
1.00E-01	1.48E-02	9.84E-01	9.99E-01
2.00E-01	2.76E-02	9.63E-01	9.91E-01
3.00E-01	3.31E-02	9.41E-01	9.74E-01
4.00E-01	3.44E-02	9.15E-01	9.49E-01
5.00E-01	3.38E-02	8.83E-01	9.17E-01
6.00E-01	3.24E-02	8.46E-01	8.79E-01
7.00E-01	3.07E-02	8.06E-01	8.37E-01
8.00E-01	2.91E-02	7.63E-01	7.92E-01
9.00E-01	2.77E-02	7.19E-01	7.47E-01
1.00E+00	2.65E-02	6.74E-01	7.01E-01
2.00E+00	2.08E-02	3.05E-01	3.26E-01
3.00E+00	1.90E-02	1.21E-01	1.40E-01
4.00E+00	1.77E-02	4.60E-02	6.37E-02
5.00E+00	1.66E-02	1.71E-02	3.37E-02
6.00E+00	1.55E-02	6.34E-03	2.19E-02
7.00E+00	1.46E-02	2.34E-03	1.69E-02
8.00E+00	1.37E-02	8.61E-04	1.46E-02
9.00E+00	1.29E-02	3.17E-04	1.32E-02
1.00E+01	1.22E-02	1.17E-04	1.23E-02
2.00E+01	7.45E-03	0.00E+00	7.45E-03
3.00E+01	5.14E-03	0.00E+00	5.14E-03
4.00E+01	3.83E-03	0.00E+00	3.83E-03
5.00E+01	3.01E-03	0.00E+00	3.01E-03
6.00E+01	2.46E-03	0.00E+00	2.46E-03
7.00E+01	2.08E-03	0.00E+00	2.08E-03
8.00E+01	1.82E-03	0.00E+00	1.82E-03
9.00E+01	1.63E-03	0.00E+00	1.63E-03
1.00E+02	1.50E-03	0.00E+00	1.50E-03
2.00E+02	1.18E-03	0.00E+00	1.18E-03
3.00E+02	1.15E-03	0.00E+00	1.15E-03
4.00E+02	1.13E-03	0.00E+00	1.13E-03
5.00E+02	1.11E-03	0.00E+00	1.11E-03
6.00E+02	1.10E-03	0.00E+00	1.10E-03
7.00E+02	1.08E-03	0.00E+00	1.08E-03
8.00E+02	1.07E-03	0.00E+00	1.07E-03
9.00E+02	1.05E-03	0.00E+00	1.05E-03
1.00E+03	1.03E-03	0.00E+00	1.03E-03
2.00E+03	8.92E-04	0.00E+00	8.92E-04
3.00E+03	7.72E-04	0.00E+00	7.72E-04
4.00E+03	6.69E-04	0.00E+00	6.69E-04
5.00E+03	5.80E-04	0.00E+00	5.80E-04
6.00E+03	5.04E-04	0.00E+00	5.04E-04
7.00E+03	4.38E-04	0.00E+00	4.38E-04
8.00E+03	3.81E-04	0.00E+00	3.81E-04
9.00E+03	3.31E-04	0.00E+00	3.31E-04
1.00E+04	2.88E-04	0.00E+00	2.88E-04
2.00E+04	7.19E-05	0.00E+00	7.19E-05

Table B.2 (continued)

Ingestion	$f_1=5.00E-02$	Half-life = $1.16E+12$ Days	^{238}U	
Time After Single Intake	Fraction of Initial Intake In:			
(Days)	Systemic		Non-Systemic	
	24-Hour Excreta	Accumulated Excreta	24-Hour Feces	Accumulated Feces
1.00E-01		7.60E-04		6.06E-04
2.00E-01		3.32E-03		5.83E-03
3.00E-01		6.71E-03		1.90E-02
4.00E-01		1.02E-02		4.07E-02
5.00E-01		1.33E-02		6.99E-02
6.00E-01		1.61E-02		1.05E-01
7.00E-01		1.85E-02		1.45E-01
8.00E-01		2.04E-02		1.87E-01
9.00E-01		2.20E-02		2.31E-01
1.00E+00	2.34E-02	2.34E-02	2.76E-01	2.76E-01
2.00E+00	5.77E-03	2.92E-02	3.69E-01	6.45E-01
3.00E+00	1.83E-03	3.10E-02	1.84E-01	8.29E-01
4.00E+00	1.30E-03	3.23E-02	7.52E-02	9.04E-01
5.00E+00	1.15E-03	3.34E-02	2.89E-02	9.33E-01
6.00E+00	1.04E-03	3.45E-02	1.08E-02	9.44E-01
7.00E+00	9.49E-04	3.54E-02	4.00E-03	9.48E-01
8.00E+00	8.66E-04	3.63E-02	1.48E-03	9.49E-01
9.00E+00	7.92E-04	3.71E-02	5.44E-04	9.50E-01
1.00E+01	7.25E-04	3.78E-02	2.00E-04	9.50E-01
2.00E+01	3.27E-04	4.25E-02	0.00E+00	9.50E-01
3.00E+01	1.73E-04	4.49E-02	0.00E+00	9.50E-01
4.00E+01	1.04E-04	4.62E-02	0.00E+00	9.50E-01
5.00E+01	6.76E-05	4.70E-02	0.00E+00	9.50E-01
6.00E+01	4.61E-05	4.75E-02	0.00E+00	9.50E-01
7.00E+01	3.20E-05	4.79E-02	0.00E+00	9.50E-01
8.00E+01	2.25E-05	4.82E-02	0.00E+00	9.50E-01
9.00E+01	1.59E-05	4.84E-02	0.00E+00	9.50E-01
1.00E+02	1.13E-05	4.85E-02	0.00E+00	9.50E-01
2.00E+02	5.24E-07	4.88E-02	0.00E+00	9.50E-01
3.00E+02	1.85E-07	4.89E-02	0.00E+00	9.50E-01
4.00E+02	1.71E-07	4.89E-02	0.00E+00	9.50E-01
5.00E+02	1.68E-07	4.89E-02	0.00E+00	9.50E-01
6.00E+02	1.65E-07	4.89E-02	0.00E+00	9.50E-01
7.00E+02	1.62E-07	4.89E-02	0.00E+00	9.50E-01
8.00E+02	1.59E-07	4.89E-02	0.00E+00	9.50E-01
9.00E+02	1.57E-07	4.90E-02	0.00E+00	9.50E-01
1.00E+03	1.54E-07	4.90E-02	0.00E+00	9.50E-01
2.00E+03	1.30E-07	4.91E-02	0.00E+00	9.50E-01
3.00E+03	1.11E-07	4.92E-02	0.00E+00	9.50E-01
4.00E+03	9.54E-08	4.93E-02	0.00E+00	9.50E-01
5.00E+03	8.21E-08	4.94E-02	0.00E+00	9.50E-01
7.00E+03	6.14E-08	4.96E-02	0.00E+00	9.50E-01
8.00E+03	5.32E-08	4.96E-02	0.00E+00	9.50E-01
9.00E+03	4.62E-08	4.97E-02	0.00E+00	9.50E-01
1.00E+04	4.01E-08	4.97E-02	0.00E+00	9.50E-01

Fig. B.1. Urinary Uranium Excretion Function -- Following Acute Intake

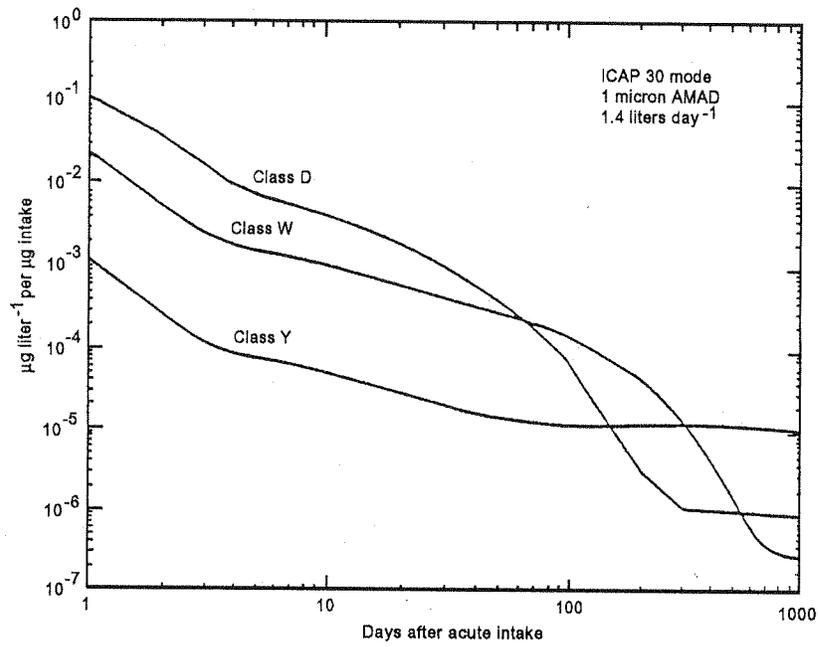
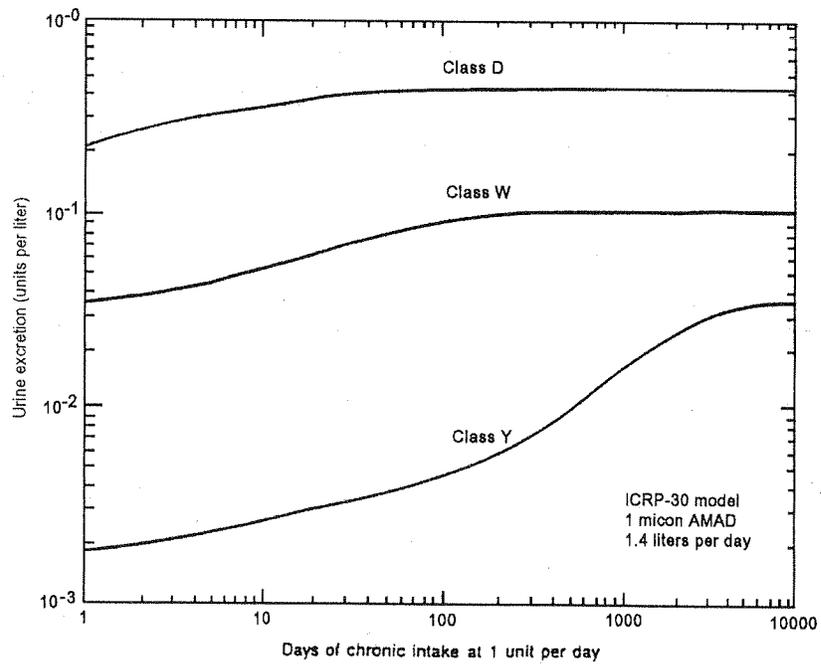


Fig. B.2. Urinary Uranium Excretion of Chronically Inhaled Uranium



Appendix C

Analytical Methods and Procedures

(This Appendix is not a part of American National Standard HPS N13.22-1995.)

C.1 Collection of Samples

C.1.1 Collection of Urine Samples

Urine samples should be collected in plastic bottles that minimize wall adsorption, and should be preserved with several drops of concentrated hydrochloric acid as soon as delivered to the plant health physicist or laboratory. The sample should be checked to see that all necessary identification information – date of sample, date of last previous void, and other information as necessary for sample evaluation and interpretation of results – are recorded both on the sample container and in the permanent sample log.

Each sampling interval may include both a Friday and a Monday sample, each maintained for separate analysis, to allow assessment of the degree of transportability of the uranium and the degree of chronicity of the internal burden of uranium. These data will be needed in the more accurate assessment of any larger exposures that might occur. Also, the additional samples allow for detection of any unusual external contamination of samples, which can be expected to occur with appreciable frequency in the industrial environment.

The Friday sample should, where possible, consist of a cumulation of all urines excreted during the work day, from an early morning sample taken upon arrival and before work, to samples taken after washing and before eating or coffee breaks, and after showering and before leaving for home. These samples will be submitted at a special station on the "clean" side of the change room that is maintained for this purpose, and that is routinely monitored and kept free of removable uranium contamination (less than 100 alpha dpm per 100 cm²). The workers should be provided with clean, disposable plastic gloves, to be worn after washing hands and prior to collecting urine samples. Workers should be instructed by the plant health physicist on the need for caution in carrying out these and other procedures for minimizing urine sample contamination.

The Monday morning sample should be collected either at home or in the clean area designated for urine collections, before the worker enters any areas where there is airborne uranium or surface contamination.

Whenever special bioassays are to be made, to confirm or follow up analyses as required by the actions and action levels of Tables 2 to 7, the samples may be taken for each void and analyzed separately, since urinary excretion following acute intakes can be expected to vary rapidly for the first hours and days following exposure (Fisher et al. 1990, 1991; Fisher 1986; Bernard and Struxness 1957; Chase 1988). The examination of this rapid transient pattern can be valuable in early assessments of the severity of the exposure, and the possibility of transient or permanent kidney damage. It can also be helpful to the plant physician as one of the indicators for determining the need for immediate bicarbonate administration, or possibly decorporation therapy (NCRP 1980). Thus, whenever air concentrations or other information indicate an acute, accidental intake, an immediate urine and fecal sampling program should begin. Excreta samples should be taken as soon as initial worker nasal and sputum sampling, contamination monitoring, and decontamination has been completed (within the first 2 h, preferably, to follow the transient removal of uranium from the lung) and should be followed by *in-vivo* counting using the chest counting equipment. In such special bioassays, the urine sample is a major part of the sequential data bank to be used in assessing intakes and internal exposures, and special care should again be taken to avoid contamination of these samples from external sources.

C.1.2 *In-vivo* Measurements

The plant health physicist should arrange the scheduling of *in-vivo* counting. Any plant *in-vivo* counting system should be installed in an area remote and/or adequately isolated from plant operations with unsealed uranium. Special bioassays resulting from acute accidental intakes or other action levels in Tables 2 to 7 should include immediate *in-vivo* counting at the plant, followed by detailed *in-vivo* counting as part of an overall medical and health physics follow-up at a specialized medical health physics center devoted to the evaluation and management of internal radiation exposure cases (NCRP 1980; Brodsky et al. 1968). Facilities not having in-house *in-vivo* capability may want to have formal written arrangements with the appropriate center(s) and procedures for referring workers to such centers as part of the special bioassay program.

The *in-vivo* counting system to be maintained at the plant, under the supervision of the plant health physicist, should have a detection capability [MDA at 5% probability of Type I or Type II errors (Brodsky 1986)] of 9 nCi in the lung, as provided for in Regulatory Guide 8.22 (U.S. NRC 1988) for a subject counting time no longer than 40 min.

The specialized medical health physics center should be one that has an MDA at least as low as 0.2 nCi for ²³⁵U in the lung, consistent with the MDA in the Health Physics Society Bioassay Performance Standard (1987).

C.1.3 Collection of Fecal Samples

Fecal samples should be collected in contamination-free areas at work or at home, as feasible and applicable to the special or routine bioassay program in effect for the particular incidental exposures, or as requested by the plant health physicist to accompany *in-vivo* measurements of Type Y material depositions in the lung.

Convenient collection devices (such as portable camping toilets) may be provided, with wide-mouth containers that can be easily sealed. No preservative is necessary for fecal samples, but they should be refrigerated as soon as received by the plant health physics staff, who should record all pertinent data and check that the sample is properly labelled. The worker will be instructed to avoid contaminating each fecal sample with urine, and to collect the urine separately at about the same time as the fecal sample, if possible.

Additional procedures for collecting the fecal sample should be consistent with provisions of the Health Physics Society standard (1987).

C.2 Overview of Analytical Methods for Uranium Determination

C.2.1 Summary of Urinalysis Methods According to Type of Exposure

C.2.1.1 Class D or W Exposure

A fluorimetric procedure may be used for urine assay of mass quantities of uranium. This procedure should have an MDA of 5 $\mu\text{g L}^{-1}$ or less, as provided by Regulatory Guide 8.22 and the Health Physics Society (HPS) standard (1987; U.S. NRC 1988). The procedure should be carried out with duplicate aliquots of at least 0.2 mL each, with dual counting of each aliquot for special bioassays. Additional aliquants should be analyzed when duplicate results are not within limits of the HPS standard. MDAs should be checked for fluorimetric procedures using formulations such as those in Appendix D.

Many Department of Energy DOE facilities now use alpha spectrometry to meet recommendations of ANSI N13.30 for enriched uranium, high in ²³⁴U, which cannot be measured adequately with fluorometric mass techniques. Inductively-coupled plasma (ICP) mass spectrometry is now commonly used for mass determinations.

For special (or diagnostic) bioassay follow-up measurements, additional aliquots of each sample should be sent to a reputable laboratory capable of reliable quantitative analyses of uranium according to one of the quantitative methods described by Wessman (1984). Additional analytical methods have been presented by Singh (1994).

C.2.1.2 Class Y Exposures

Sometimes Class Y exposures include a small proportion of more transportable uranium compounds that can be detected in the urine during the early days after exposure. Additional sensitivity for low enrichments or natural uranium may also be provided at the plant by use of a laser phosphorimeter, now available commercially with a detection level of about 0.5 mg L^{-1} or less (Miller 1983). To separate uranium chemically, a rapid liquid ion exchange method may be carried out with at least 100 mL of urine (Butler et al. 1966; Butler 1968). This will allow early detection of important sources of exposure at the plant that might include inhalation of Class Y materials if urinalysis is scheduled together with *in-vivo* counting soon after suspected exposures.

C.2.1.3 Special Bioassay Follow Up

Situations requiring special bioassay confirmations should activate a program of confirmatory analyses of high accuracy and yield using a contracted laboratory capable of carrying out radiochemical procedures of the type described by Wessman (1984).

C.2.2 In-vivo Counting

The counting procedures at the plant and at the medical health physics center will be appropriate to the systems described in Section C.1.2. Procedures for system calibration, phantom blank measurements, worker decontamination prior to measurement, and quality assurance can be selected from those provided in the Health Physics Society standard for *in-vivo* counting systems (Health Physics Society 1987). Counting methods and data are reviewed in some detail in Section C.4.4.

C.2.3 Fecal Analysis

For rapid initial measurements of uranium in feces at the plant or service laboratory, fecal samples may be counted without radiochemical separations to save time and avoid problems with radiochemical yield. Counting systems, such as the Phoswich detectors of Guilmette (1986) or Kramer et al. (1987), can be used with minimum wet ashing of the sample in concentrated nitric acid when necessary to reduce sample size and volume to fit the well counter and reduce self-absorption of low-energy x rays.

For special bioassay follow up, aliquots of about 50 g of feces from each sample will be sent to a laboratory with capabilities for high-performance uranium analyses such as those described by Wessman for solid samples (Wessman 1984). Appropriately spiked simulated fecal samples (ICRP 1975) and blank simulated fecal samples must also be included with each batch of fecal samples analyzed, either at the plant or at an outside laboratory.

Since routine fecal analysis requires for ultimate sensitivity and accuracy the detailed radiochemical procedures best carried out by a radiochemist experienced with the analysis, aliquants of fecal samples in significant exposure cases will be sent, along with appropriate spiked samples and controls, to a reputable analytical laboratory for analysis. A simulated fecal material such as that described by Kramer et al. (1987) may be used for making spiked samples with known quantities of enriched uranium in the range 10–1,000 dpm alpha activity for use in checking the accuracy of contracted laboratory work as well as in-plant analyses. Also, spiked samples, as well as the higher-activity excreted samples from each case, can be measured in the Phoswich-type detector for comparison with the more detailed radiochemical analyses to check that quantitative yields from the radiochemical analyses of actual fecal samples are high and consistent. Only a most rigorous attention to proper radiochemical procedures will ensure that recoveries of

metabolized uranium in fecal material will be consistent with recoveries of spiked uranium in simulated matrices (Wessman 1984; ICRP 1975). In addition, any ashing of blanks with nitric acid should be carried out with the same simulated fecal material and reagents as used for the corresponding spiked "standards" to be counted in the Phoswich detector system.

C.3. In-Plant Facilities, Equipment, and Supplies Needed for Analytical Procedures

For some plants processing larger quantities of uranium (Brodsky 1980, 1989; McGuire 1985), supplies, equipment, and laboratory facilities should be available at the plant for diagnostic assessments related to bioassay during the first 1 to 2 d after an acute intake. Experience shows that early first aid treatment, decontamination, and initial bioassay and internal dose assessments may need to be done at the plant the first day or two after a serious accidental release, even when firm arrangements for medical treatment and follow up have been made in advance at a medical health physics evaluation center (MHEC) (Fisher et al. 1990; Wagner 1994; Moore and Kathren 1985; NCRP 1980; Brodsky et al. 1968). Even after contaminated persons have been transferred to an MHEC, certain analytical work and evaluations might need to be carried out by the plant health physics team.

Analytical laboratory space will be needed at the plant for preparation of spiked QA control samples to test the analytical service laboratory. Space will also need to be designated for early analysis of aliquots of bioassay samples, as well as samples from air filters in the vicinity of accidents or any other samples that will help to characterize the nature of aerosols inhaled. Any air or other environmental samples that may have been taken in the vicinity of those who have been exposed, either off-site or on-site, should be immediately collected and analyzed as soon as possible. Chemical, particle size, and radiometric analysis methods should be employed to characterize as well as quantitate the airborne material to which persons have been exposed.

In-plant analytical laboratories for use in routine bioassay or in emergencies will need to be located or protected so that they cannot be contaminated. Situations have occurred where the plant analytical equipment was contaminated by the incident itself, preventing early radiochemical analysis of the type of dust involved in the incident. This type of situation can force the MHEC team to use assumptions that grossly overestimate intakes for the first several days after an accident.

C.4 Specific Information on Selected Analytical Procedures

C.4.1 Fluorimetric Analyses

Fluorimetric procedures for emergency screening of urine and feces immediately after single intakes may have adequate detection capabilities if the minimum detectable amounts (MDAs) are $25 \mu\text{g L}^{-1}$ for urine, or $25 \mu\text{g}$ per sample for feces or other biological samples. However, follow-up measurements and routine urine analyses should be carried out by procedures that meet, as a minimum, the acceptable MDAs of the HPS standard (1987): $5 \mu\text{g L}^{-1}$ for urine or $5 \mu\text{g}$ per sample for feces.

Fluorimetric analyses meeting the screening detection capability can be designed from routine methods of analysis so that they can be performed within 3-4 h (Sedlet et al. 1965; Gautier 1993; Sedlet 1965). This includes drying carefully to a red heat to remove carbon, leaving the few salts in the urine, then melting the residue in a flux of sodium fluoride, cooling and reading on a fluorimeter. This method should provide sufficient detection capability for emergency analyses. Sedlet et al. (1965) suggest that uranium is best measured in emergencies by fluorescence.

With older instruments, in which uranium is measured after fusing a 0.2-mL sample of urine with sodium fluoride, Sedlet suggests that the procedure can be carried out in 30 min, with a detection capability of $4 \mu\text{g}$ of uranium per 1,400 mL of urine. Since the alpha activity of natural uranium is 6.77×10^{-7} pCi (U.S. NRC

1988), this is equivalent to 2.7 pCi of natural uranium or 270 pCi of enriched uranium (1% ^{234}U) per 1,400 mL (extrapolated up from the 0.2-mL aliquot measured). This detection capability is approximately consistent with the acceptable MDA often obtained in routine urinalysis for uranium, considering the different methods of estimating "detection limits" and MDAs (HPS 1987; Brodsky 1986; Sedlet et al. 1965; Sedlet 1965).

C.4.2 Radiochemical Separations Followed by Radiometric Analysis

For analytical procedures completed by alpha radiometric measurements, the corresponding MDAs for emergency screening at the plant site should be 0.5 pCi L⁻¹ or per sample; the follow-up measurements at the analytical laboratory or MHEC should have MDAs of 3.7 Bq m⁻³ (0.1 pCi L⁻¹) or per sample or less (HPS 1987).

Uranium can be separated from urine fairly rapidly by extraction into di(2-ethylhexyl)-phosphoric acid (HDEHP) dissolved in benzene (Sedlet et al. 1965; Sedlet 1965). Then the extract can be counted directly in a liquid scintillation counter or evaporated to dryness and counted in an alpha proportional counter (Sedlet et al. 1965; Sedlet 1965). Butler has provided procedures for exchanging uranium to either of the liquid ion exchangers HDEHP or tri-isooctylamine (TIOA) (Butler et al. 1966).

An improved procedure by Butler (1968) yields low amounts of solids, with more than 95% recovery of uranium and no other radioactive material. This procedure uses only 100 mL of urine and takes only 4 h. The time for electrodeposition is eliminated. The residue can be mounted on special 3/4-inch stainless steel planchets (Butler 1968) and counted with 30% efficiency in alpha spectrometers having five counts per day background. Sensitivity would be 0.25 dpm alpha per 1.5 L of urine for a 100-mL sample of urine. This would be more than adequate for emergency use and quite satisfactory for routine determinations of enriched uranium in urine at the action levels of Tables 2 and 7 (Section 6).

Royster has described an electrodeposition procedure, preceded by a step that will eliminate calcium, which is fairly simple and requires only 1 h of electrodeposition to obtain yields of about $90 \pm 4\%$ (Royster 1960). Patterson described a method of electrodeposition from "raw" urine that takes even less time for electrodeposition (Patterson 1958).

Patterson's procedure should be done in duplicate on each urine. Spikes may be added to one of the duplicate samples using uranium standards made from standard reference materials obtained from the National Institute of Standards and Technology (Inn and Noyce 1982). In this way, the somewhat variable yields may be checked for adequate accuracy by comparing the results from the two aliquots.

Uranium can also be co-precipitated for alpha spectroscopy, with the entire procedure from sample dissolution to the final fractions ready for coprecipitation on cerium hydroxide or fluoride taking about 5 h for eight samples (Sill 1981; Bernabee 1983).

Sedlet (1965; Sedlet et al. 1965) recommends sample sizes of no more than 50-100 mL of urine if analytical procedures are to be kept to 1 h or less. Fecal samples should be kept to 10-20 g (wet weight) and blood samples to 10 mL. Fecal and blood samples will need to be ashed first to reduce organic matter and then redissolved, which will take additional analytical time of at least several hours (Sedlet et al. 1965; Sedlet 1965).

However, Sill et al. (1964) emphasize caution on two important points in the chemical analysis of feces (or any other materials that might contain large amounts of calcium). First, the addition of sulfuric acid or sulfates should be avoided because the calcium sulfate formed will occlude tri- and quadrivalent elements with ionic radii greater than 1 Angstrom, which can include other radioactive elements such as thorium, rare earths, and the transuranium elements. Second, samples should not be dry-ashed or even allowed to completely dry and bake during wet ashing unless the residue is to be fused with pyrophosphate. Otherwise, the phosphates present are converted to a white solid that is extremely insoluble even in concentrated acids.

These white solids will contain most of the components present in the sample and would thus deter a rapid analysis (Sill et al. 1964).

C.4.3 Low-Energy-Photon Counting of Bulk Excreta Samples

Guilmette (1986) reported on the construction and use of a low-energy photon detector for radioassay of plutonium or americium in biological samples. This same detector system should be readily adaptable to measurements of uranium in bulk samples, especially for diagnostic bioassays with known accidental intakes. Lower energy photons (less than about 100 keV) from the uranium isotopes and daughters ^{234}Th and ^{231}Th can be used for detection. Also, Guilmette's design was fabricated commercially. Therefore, it may be available to others who need a simple and rapid system for counting samples taken in the early emergency period and who may have capabilities for only the simplest chemical reduction of biological samples. It may be available for those who wish to obtain initial estimates of intake from raw, unprocessed samples.

The detector is based on the dual-crystal, anticoincidence arrangement of Laurer and Eisenbud (1968), where the lower energy photons (and the photoelectrons they produce) are absorbed in a thin crystal of NaI(Tl). However, many of the higher energy electrons produced by other background sources penetrate both the NaI(Tl) and adjacent CsI(Tl) and produce different pulse rise times, so that many of the background-contributed pulses can be eliminated by electronic anti-coincidence circuitry, such as that described by Guilmette (1986). Guilmette's system was developed for plutonium and americium counting but could be adapted to uranium since some isotopes of uranium also emit low-energy radiation.

C.4.4 *In-vivo* Counting Methods

C.4.4.1 Conditions for Which *In-Vivo* Bioassay is Appropriate

In-vivo chest counting equipment is helpful for detection of relatively high intakes of Class Y uranium. While urinalysis is adequate for detecting exposures to the more soluble uranium compounds such as uranyl fluoride or uranyl nitrate, *in-vivo* counting can also be helpful in providing estimates of intakes of Class D and W uranium after significant intakes. The variability of intake estimates from urinary excretion analyses alone, which include variations in amounts excreted from day to day, can result in variations of up to a factor of two or more in early estimates of intake of uranium, even by the most informed and equipped centers (Brodsky et al. 1968; Sill et al. 1964).

Early measurements of the mass deposited in the lung and in other parts of the body during the first 1-2 d, combined with measurements and material balances of the total amounts eliminated in urine and feces, can be very helpful in establishing much more accurately the initial exposures in certain acute exposure situations in terms of both intake and deposition. This approach is preferable to reliance on later modelling of the (very variable) urinary excretion rates alone.

Since expensive *in-vivo* counting equipment might not be required for the routine bioassay program, for which urinalysis will ordinarily meet requirements except for the most refractory (insoluble in tissue fluids) uranium compounds, the facility may wish to make arrangements for emergency *in-vivo* counting with a commercial firm at some nominal and reasonable rate.

C.4.4.2 Discussion of *In-vivo* Measurements and Parameters

Prior to a discussion of the capabilities and limitations of *in-vivo* measurements, a description and understanding of typical uranium isotopic compositions that might be encountered is appropriate. Prior to enrichment (normal uranium), uranium is composed of 99.2765% ^{238}U , 0.0070% ^{236}U , 0.7110% ^{235}U , and 0.0055% ^{234}U (weight percent). In the ^{235}U gaseous diffusion enrichment process, in addition to the increase in ^{235}U , the proportions of ^{234}U and ^{236}U also increase. Table C.1 gives the total activity per microgram of uranium for each uranium isotope as a function of weight per cent ^{235}U for certain mixtures of isotopes. Note that for normal uranium, approximately 50% of the activity is due to the decay of ^{234}U . For enriched uranium,

the major contributor to total activity is ^{234}U . Thus, the major contributor to the dose from enriched uranium is from ^{234}U alpha. The average alpha energy for each isotope is also given in Table C.1. To be technically correct, in order to calculate dose one needs to know the isotopic distribution of the exposure material, which can vary even for the same ^{235}U enrichment (Manninen 1994).

In-vivo monitoring for normal and enriched uranium usually involves the detection of the 185.7-keV gamma emitted with an abundance of 54% from the decay of ^{235}U . Thus, uranium *in-vivo* monitoring is based on the detection and measurement of a uranium isotope that contributes very little to the dose. The data in Table C.1 are primarily for illustrative purposes and should not be used to estimate total uranium activity based on ^{235}U enrichment and the amount of ^{235}U detected by *in-vivo* measurements. Typically, limits of detection are reported in units of ^{235}U . Therefore, one must know either the total uranium activity or calculate this activity from the uranium isotopic distribution in order to determine the minimum detectable activity for any exposure material. The measurement of normal or depleted uranium can be accomplished through the detection of the 92.38- and 92.80-keV K x rays emitted by ^{234}Th (2.7% abundance each). If exposure is to freshly purified uranium or to uranium that has undergone some physical process such as melting, the ^{234}Th will not be in equilibrium with ^{238}U . Such a situation could result in an underestimate or overestimate of the actual burden, depending on the nature of the exposure.

There are currently three detector systems (1991) in use to measure lung deposition of uranium: large area and thick NaI detectors (Cofield 1966); dual NaI and CsI (Laurer and Eisenbud 1968); and high-purity germanium detectors. None of the detector systems appears to be markedly superior to the others, although somewhat better detection capability has been reported for germanium detectors (Palmer et al. 1987; Toohey et al. 1991). The MDA attainable is limited by the high count rate in the uranium region from Compton scatter of ^{40}K photons in the body of the subject being measured, and the high count rate in the 186-keV region due to the ambient background. A review of uranium *in-vivo* monitoring is included in Scott and West (1975).

Since the initial amount of uranium in the lung indicative of an internal dose exceeding any dose limits would be at least about 740 Bq (20 nCi) (and much higher for soluble forms of uranium such as uranyl fluoride), and the acceptable MDA in Regulatory Guide 8.22 for counting natural uranium in the lung under chronic exposure conditions is 333 Bq (9 nCi), it would seem feasible to devise inexpensive lung and whole-body counting systems for use in emergency screening for uranium exposure in the first few hours after an inhalation incident. The system used for chest counting by West et al. (1979) consisted of wide area counters placed near the lungs of subjects within a shielded room. Specifications and data from the earlier detection system of West et al., as given by Cofield (1966), are presented below since they were obtained using a single 9-inch diameter by 4-inch thick NaI(Tl) crystal, shielded except for the front face by a one-half-inch-thick epoxy plastic loaded with tungsten. The detector was mounted from the ceiling within an 8-inch-thick iron room. The parameters of this system are provided in some detail in Tables C.2 and C.3 below, adapted from Cofield (1966), so the detection capabilities for planned systems can be estimated by simply increasing appropriately the background counts due to background radiation, and adjusting the expected control and calibration counts (Brodsky 1986).

Although uranium lung counts have historically been based on the 186-keV gamma from ^{235}U and the 92-keV gamma associated with ^{238}U daughters, more recent compilations of spectral information on uranium and its daughters could possibly be used (Kocher 1981) for more efficient monitoring. While Cofield (1966) assumed 0.8 abundance of 186-keV gammas per alpha, more recent data indicate that the 185.720-keV gamma ray has an abundance of only 54% (Kocher 1981). However, gammas 19-27, 163.35- 205.31 keV, have a total abundance of 0.6539 per disintegration of ^{235}U , most of which will produce counts within the 50-keV wide photopeak region about the 186-keV peak, as used by Cofield (1966). The data and detection capabilities of Cofield are presented in Tables C.2 and C.3. The K-alpha and K-beta x rays of 89.95-105 keV from ^{235}U have a total abundance of 9.2% (Kocher 1981), so it could be efficacious to utilize a thinner crystal of NaI to reduce background and enhance sensitivity for these photons together with the 90-keV photons from ^{231}Th and ^{234}Th . ^{231}Th (25.52 h) will be in equilibrium with ^{235}U within days after uranium separation, so its 11.3% abundant gammas in the 81.2-99.3 keV region can be used as the "90-keV" spectrum. Likewise, its 26.6-

keV, 18.7% abundant gamma can be considered for additional sensitivity and for discriminating against skin contamination and estimating effective depths of lung or body distributions of ^{235}U . The ^{234}Th daughter (24.10 d) of ^{238}U will be in equilibrium with ^{238}U months after separation, and its 2.72% abundant 92.380-keV gamma and 2.69% abundant 92.800-keV gamma can then be used to enhance detection capabilities for natural uranium.

**Table C.1 –
Specific Activities of Each Uranium
Isotope as a Function of Weight Percent ^{235}U
and Average Alpha Energies for Each Isotope**

Percent ^{235}U	dpm/mg of Total Uranium ^a				Total
	^{234}U	^{235}U	^{236}U	^{238}U	
Normal	.745	.033	.001	.745	1.524
2%	1.514	.095	.002	.725	2.336
4%	3.854	.190	.004	.710	4.758
93%	144.52	4.427	.294	.041	149.28
Average Alpha Energy	4.76	4.26	4.48	4.18	

^a Note that exact proportions of these isotopes can vary for different enrichment processes (Manninen 1994).
1 Bq = 60 dpm.

**Table C.2 –
Parameters for Uranium *In-Vivo* Chest Counting
[as adapted from Cofield (1966)]**

Parameter	Uranium Enrichment Level		
	Level 1 Uranium Enriched 93% ²³⁵ U	Level 2 Uranium Natural 0.7% ²³⁵ U	Level 3 Uranium Depleted 0.22% ²³⁵ U
Maximum permissible body burden (mg total uranium) (MPBB)(ICRP 1960)	0.275 mg	25.615 mg	49.624 mg
MPBB content of ²³⁵ U present (mg)	0.255 mg	0.180 mg	0.109 mg
186-keV gammas per second per MPBB	16.1	11.4	6.9
Uranium calibration factors using C. E. Miller phantom (Cofield 1966) $\mu\text{g U}/(\text{count}/\text{min})$ 186 keV channel (165-215 keV)	10	935	1613
$\mu\text{g } ^{235}\text{U}/(\text{count}/\text{min})$, 186 keV ch.	9.3	6.5	3.6
$\mu\text{g U}$ per count/min, 90 keV ch. (65-115 keV)	–	4440	3840
Gross count-rate of subject with 1 MPBB of Level 1 Uranium	237 c/min 186 keV band $\pm 9.6\%$ (for 10 min count)		
Instrument background	90 $\pm 2.6\%$ (for 50 min count)		
Predicted unexposed control subject (Cofield 1966)	120 $\pm 11\%$ (for 30 min count)		
Net ²³⁵ U count-rate (237-90-120)	27 $\pm 13.1\%$ (for 30 min count)		
Error in measurement of uranium lung burden for 1 MPBB in lung (95% C.L., i.e., 2s)	$\pm 131 \mu\text{g}$ $\pm 47.6\%$ MPBB	$\pm 12,248 \mu\text{g}$ $\pm 47.8\%$ MPBB	$\pm 21,138 \mu\text{g}$ $\pm 42.6\%$ MPBB

Table C.3 – Calibration Correction Factors for Chest Thickness
[to be Divided into the 186-keV Calibration Factors
of Table C.2, as adapted from Cofield (1966)]

Chest Thickness (inches)	7.0	7.5	8.0	8.5	9.0	9.5	10.0
Cal. Correction Factor	1.19	1.10	1.00	0.90	0.81	0.73	0.65

Appendix D

Specification of Minimum Detectable Amount (MDA)

(This Appendix is not part of American National Standard HPS N13.22-1995.)

For purposes of this guide, the Minimum Detectable Amount (MDA) is defined as the smallest concentration of radioactive material in urine that has a 95% probability (chance) of being detected when measurement procedures are set so that the concentration level at or above which the decision is made that something is detected (decision limit) produces only a 5% or less, chance of calling a blank reading a positive sample (Brodsky 1986). Radioactive material is then called "detected" when the value obtained from an instrument reading is above the decision limit (DL), and is thus high enough so that it can be concluded that activity above the system background is probably present.

For a fluorimetric measurement where the "blank" urines fluctuate with a standard deviation, s_b , the DL corresponds to an amount that will on average give a reading of about $2.33s_b$. For a radiometric count, the DL corresponds to an activity that gives a total count equal to $2.33s_b$, where s_b is the standard error in the total count. This standard error should take into account variations in readings due to variations in analytical processes as well as poison counting fluctuations. This total variation can be determined only by replicate determinations on appropriate blank samples.

The MDA is defined *more conservatively* than the DL so that if an amount of uranium equal to the MDA is present above background, it will have a 95% probability (chance) of being detected (Brodsky 1986). A review of concepts and derivations of DL and MDA, including Currie's original derivations, are included in Brodsky 1986. For a particular radiochemical separation and fluorimetric measurement process (which may include a radiochemical separation for a uranium urinalysis procedure), the MDA is defined as follows:

$$\text{MDA} = \frac{4.65 s_b}{\text{KEVY (v/V)}} \quad (\text{D.1})$$

where

- MDA = the minimum detectable amount (or concentration);
- s_b = the standard deviation of fluctuations in fluorimeter blank measurements, or count rate (counts per second), for a specific time of measurement and specific aliquot volume;
- K = conversion or calibration factor to convert units of s_b from instrument scale reading units to mass or activity units; units of K may be microamps per microgram, or 3.7×10^4 d per second- μCi if activity is counted to obtain the final result (this term is omitted if s_b is given in microcuries directly by use of a calibration standard);
- E = the counting efficiency (counts per disintegration), or = 1 when a fluorimetric standard is measured in the same geometry as the sample;
- V = sample volume (L) of total urine sample;
- v = volume of aliquot taken from the urine sample and added to the flux in the fusion dish (L); and
- Y = the fractional radiochemical yield or recovery (if applicable).

Examples of the calculation of MDA are given in Brodsky (1986).

Generalized and Simplified Formula for MDA for Radiometric Determinations

When final determinations require a radiometric counting procedure, the most general formulation of MDA (and perhaps the simplest to understand) is:

$$\text{MDA} = \frac{4.65 s_b + 3}{KT}, \quad (\text{D.2})$$

where

S_b = standard deviation of total counts of replicated blanks;

K = "calibration constant," (e.g., counts per minute per Bq), obtained from analyses and count measurements of appropriate spiked samples;

T = total counting time, in minutes, for each replicated blank (and for each sample to be determined in a specified analytical procedure); and

3 = constant to provide appropriate Type II error for blank counts less than 25 in T minutes, or for "low-background" determinations.

It must be noted, however, that this formula provides an MDA representing the amount having only 0.05 chance of going undetected, when a decision that a *net* count is positive is made at or above the decision level (DL):

$$\text{DL} = 2.33 S_b \quad (\text{D.3})$$

This DL provides a 0.05 probability of a "false positive" (Type I error) under conditions when the blank count is high enough so that its distribution (and the distribution is net count of a zero activity sample) is approximately normal (Brodsky 1986).

When blank counts are low (below 25) in the spectral region(s) of interest, the probabilities of Type I and Type II errors, using eqns (D.3) and (D.2), respectively, can differ somewhat from 0.05. In such cases, specification of Type I or Type II errors, or modification of MDA formulations, if warranted, can be done using exact Poisson probability distributions for the blank and sample counts (Brodsky 1992). Strom and McGuire (1993) discuss additional ways of improving detection capabilities.

Since, in eqn (D.2), $S_b = \sqrt{RT}$, where R is the blank count rate, the MDA can be decreased inversely as the \sqrt{T} when 3 in the numerator is negligible. Conversely, for low background (blank) counting, MDA decreases inversely as T when $4.65 S_b$ is small compared to 3. Thus, the effective MDA for an increased number of samples is also decreased according to the counting time, as indicated for 3 negligible or dominant, respectively (Strom and McGuire 1993; Brodsky 1986).

Eqns (D.2) and (D.3), and others for use when fixed (bias) errors are unknown but can be bounded, are presented and discussed in more detail in Brodsky 1986.

Appendix E

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(This Appendix is not a part of American National Standard HPS N13.22-1995.)

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