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Internal Dose Reconstruction

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☐ New ☒ Total Rewrite ☐ Revision

PUBLICATION RECORD

EFFECTIVE DATE	REVISION NUMBER	DESCRIPTION
02/06/2007	00	This technical information bulletin provides guidance in the performance of internal dose reconstructions. It incorporates the requirements of ORAUT-PROC-0003, *Internal Dose Reconstruction*, which will be cancelled. Incorporates internal formal and NIOSH review comments. Incorporates additional NIOSH review comments. There is no change in the assigned dose and no PER required. Training is required: As determined by the Task Manager. Initiated by Elizabeth M. Brackett.
09/08/2014	01	Revision to bring the document up to date with current practice and to provide additional guidance to dose reconstructors. Incorporates formal internal and NIOSH review comments. Constitutes a total rewrite of the document. Training is required: As determined by the Objective Manager. Initiated by Elizabeth M. Brackett.
01/09/2018	02	Revision to address comments from the Procedures Review Subcommittee and to provide additional guidance and information to the dose reconstructors on current methods. Incorporates formal internal and NIOSH review comments. Constitutes a total rewrite of the document. Training is required: As determined by the Objective Manager. Initiated by Elizabeth M. Brackett.

11/25/2025	03	Revised to reflect various updated methods for internal dose assessment that have evolved over time and to provide the technical basis for some of the guidance. Several sections were expanded and others moved to improve the flow of the document. New sections include:
		
		• How to handle various types of bioassay data.
		  ○ Invalid data (Section 3.4.1)
		  ○ Offsite data (e.g., body counts performed at a different location/facility) (Section 3.4.2)
		  ○ Positive baseline samples (Section 3.4.3)
		  ○ Post-employment results (Section 3.4.4)
		  ○ Added detail on the use of chest counts. (Section 3.4.6)
		  ○ Fecal samples (Sections 5.2.2, 6.3)
		• Assessment of special H-3 compounds. (Section 3.3.2.3)
		• Calculation of MDA from the reported error or DL when not included in a report. (Section 3.4.5)
		• Discussion on the use of internal dose assessment software. (Section 4.0)
		• Assessment of missed dose from an incident. (Section 5.2.3)
		• Assessment of intakes following chelation. (Section 10.2)
		
		Incorporates formal internal and NIOSH review comments. Constitutes a total rewrite of the document. Training is required: As determined by the Objective Manager. Initiated by Scott R. Siebert and authored by Elizabeth M. Brackett.

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ACRONYMS AND ABBREVIATIONS

AEB	Alvin E. Blackwell
AI	alveolar-interstitial region
AMAD	activity median aerodynamic diameter
AVLIS	atomic vapor laser isotopic separation
AWE	Atomic Weapons Employer
bb	bronchiolar region
BB	bronchial region
Bq	becquerel
CEP	Controls for Environmental Pollution
C.F.R.	*Code of Federal Regulations*
d	day
DL	decision level
DOE	U.S. Department of Energy
DOL	U.S. Department of Labor
dpm	disintegrations per minute
DTPA	diethylene-triamine-pentaacetic acid
DU	depleted uranium
EEOICPA	Energy Employees Occupational Illness Compensation Program Act of 2000
ET1	extrathoracic region 1
ET2	extrathoracic region 2
EU	enriched uranium
F	fast (absorption type)
g	gram
GI	gastrointestinal
GSD	geometric standard deviation
HEU	highly enriched uranium
HPS	Health Physics Society
HTO	tritiated water
ICD-10	*International Statistical Classification of Diseases and Related Health Problems, 10th Revision*
ICP-MS	inductively coupled plasma–mass spectrometry
ICRP	International Commission on Radiological Protection
IDOT	Internal Dosimetry Tool
IMBA	Integrated Modules for Bioassay Analysis
IREP	Interactive RadioEpidemiological Program
JCGM	Joint Committee for Guides in Metrology
K	special absorption type for uranium aluminide
keV	kiloelectron-volt, 1,000 electron-volts
L	liter
LEU	low-enriched uranium

LLI	lower large intestine
LN(ET)	extrathoracic lymph nodes
LN(TH)	thoracic lymph nodes
LOD	limit of detection
M	moderate (absorption type)
MDA	minimum detectable amount
MDD	minimum detectable dose
MHI	Mound highly insoluble (absorption type)
mL	milliliter
nCi	nanocurie
NCRP	National Council on Radiation Protection and Measurements
NIOSH	National Institute for Occupational Safety and Health
NM	nonmonotonic (absorption type)
OBT	organically bound tritium
ORAU	Oak Ridge Associated Universities
ORAUT	ORAU Team
PER	program evaluation report
PIDS	Principal Internal Dosimetry Scientist
POC	probability of causation
REF	radiation effectiveness factor
RU	recycled uranium
S	slow (absorption type)
SI	small intestine
SMT	stable metal tritide
SRDB Ref ID	Site Research Database Reference Identification (number)
SS	super S (absorption type)
TIB	technical information bulletin
TIMS	thermal ionization mass spectrometry
ULI	upper large intestine
U.S.C.	*United States Code*
V	vapor (absorption type)
WLM	working level month
WNA	World Nuclear Association
σ	standard deviation
µCi	microcurie
µg	microgram
µm	micrometer
§	section or sections

1.0 INTRODUCTION

Technical information bulletins (TIBs) are not official determinations made by the National Institute for Occupational Safety and Health (NIOSH) but are rather general working documents that provide historical background information and guidance to assist in the preparation of dose reconstructions at particular sites or categories of sites. They will be revised in the event additional relevant information is obtained about the affected site(s), such as changing scientific understanding of operations, processes, or procedures involving radioactive materials. TIBs may be used to assist NIOSH staff in the completion of individual dose reconstructions.

In this document the word “facility” is used to refer to an area, building, or group of buildings that served a specific purpose at a U.S. Department of Energy (DOE) or Atomic Weapons Employer (AWE) facility. It does not mean, nor should it be equated to, an “AWE facility” or a “DOE facility.” The terms AWE and DOE facility are defined in 42 *United States Code* (U.S.C.) § 7384l(5) and (12) of the Energy Employees Occupational Illness Compensation Program Act of 2000, respectively.

1.1 PURPOSE

This TIB provides information and guidance for reconstructing internal dose for the Energy Employees Occupational Illness Compensation Program Act (EEOICPA) dose reconstruction (the Project) and documents the rationale for selection of certain default parameters. It includes information that might be useful in the development of technical basis documents (e.g., discussions on mass vs. activity fractions of uranium mixtures and the relative dose conversion factors of various nuclides).

1.2 SCOPE

There are many approaches that can be taken when reconstructing the internal dose component for a claim. These options depend on several factors including the employment site, type and number of cancers, and availability of monitoring data for the worker. There might be approaches in this document that are not applicable or appropriate to a specific claim. The assessment of internal dose can be a complicated process, particularly in dose reconstruction where unmonitored and undetected doses must be included. This TIB provides as much detail as possible based on situations that have been encountered, but there might be cases that present unique or contradictory information that require a modification to these approaches. In such cases, contact the Principal Internal Dosimetry Scientist (PIDS) for assistance.

NOTE: Any methods not specifically addressed in this TIB, including those that come from guidance in consultation with the PIDS, must be documented in the dose reconstruction report.

The terminology and methods in this TIB are applicable for reconstructing doses on the Project but are not necessarily reflective of standard internal dosimetry practices in an operational program where the program is meant to meet regulatory standards. Quantities of interest are different, and the methods in this document are favorable to the claimant when a parameter is unknown rather than the typical operational approach of a “most likely” value when there is uncertainty.

This document provides default values for use only when there is no better information in the claim file or the site profile. Worker information also takes precedence over default values in the site profile. For example, most site profiles contain tables of minimum detectable amounts (MDAs) for urine sample results. If the claim file includes an explicit or implied MDA or reporting level in the bioassay result listing (e.g., “<” appears in front of a value), this specific value should be applied to the claim rather than the default value from the site profile. Note that this direction does not apply if that information is

determined to be invalid, such as in the case of laboratory results that have been demonstrated to be inaccurate.

Attachments A to D provide detailed information on lung retention, default material types, sample recounts, and radiation types by nuclide.

2.0 TERMINOLOGY AND BACKGROUND INFORMATION

This section introduces Project terminology and provides definitions and additional information necessary to completing an internal dose assessment. It also provides instructional material that might be useful for understanding some internal dosimetry concepts.

2.1 SYSTEMIC VERSUS NONSYSTEMIC

These terms are used by the International Commission on Radiological Protection (ICRP).

“Nonsystemic” organs are those in which material is directly deposited in the organ, specifically the respiratory tract through inhalation and the gastrointestinal (GI) tract through ingestion. The lymph nodes are included in the nonsystemic organs. The following regions are defined by the ICRP:

- The respiratory tract includes extrathoracic regions 1 and 2 (ET1 and ET2), lungs, thoracic lymph nodes [LN(TH)], extrathoracic lymph nodes [LN(ET)], bronchial region (BB), bronchiolar region (bb), and alveolar-interstitial region (AI) [ICRP 1994a].

- Components of the GI tract include the stomach, small intestine (SI), lower large intestine (LLI), upper large intestine (ULI), and colon (note that the colon is simply the mass weighted average of the LLI and ULI) [ICRP 1979].

“Systemic” organs and tissues are those to which radioactive material is transferred through blood circulation. Material reaches the blood through the nonsystemic organs described above, through direct absorption through the skin, or from a wound or injection site.

2.2 METABOLIC VERSUS NONMETABOLIC ORGAN

Several organs are included in the general models in ICRP Publication 78 [ICRP 1998]. However, for a given element, only a specific subset of these organs is included in the metabolic or biokinetic modeling. Organs that are specifically modeled are referred to as “metabolic” for this Project. Others for which a dose is calculated but are not specified by the ICRP element-specific model are referred to as “nonmetabolic.” The biokinetic models are based on the behavior of the specific element in the body, so the metabolic organs vary with the element of interest. For all elements, the metabolic organs include the GI tract [ICRP 1979] and, in the case of inhalation the respiratory tract [ICRP 1994a], because material is always deposited in these regions. In addition, if any part of the bone is specified in the ICRP model for an element, all bone parts, including red bone marrow and bone surface, are considered metabolic.

ORAUT-OTIB-0005, *Internal Dosimetry Organ, External Dosimetry Organ, and IREP Model Selection by ICD-10 Code* [Oak Ridge Associated Universities (ORAU) Team (ORAUT) 2019], correlates the codes in *The International Statistical Classification of Diseases and Related Health Problems, 10th Revision* (ICD-10) [World Health Organization 2016] with the appropriate organs and tissues to be modeled in the Integrated Modules for Bioassay Assessment (IMBA) and the Internal Dosimetry Tool (IDOT) computer programs. For claims where the organ of interest is not included in IMBA or IDOT, the use of “highest nonmetabolic organ” is specified. In such situations, the dose to be assigned to the organ is the largest dose among the reported organs that are not part of the ICRP biokinetic model for

the particular radionuclide. See Section 3.1.1 of OCAS-IG-002, *Internal Dose Reconstruction Implementation Guideline*, for additional information on this determination (note that the numbering in the document is out of order and Section 3.1.1 falls under Section 3.3) [NIOSH 2002].

2.3 ABSORPTION TYPES

Absorption type, also referred to as material type, describes the rate of absorption from the *respiratory tract* into blood. ICRP Publication 66, which specifies the human respiratory tract model, describes three types: F (fast solubilization), M (moderate solubilization), and S (slow solubilization) [ICRP 1994a]. Once material reaches the blood, all absorption types behave according to the biokinetic model for the element. Attachment A contains additional information about the behavior of material in the lungs.

Note that absorption type does *not* indicate how long material is retained in the *total body*. Once removed from the lungs, material is transported to the systemic organs. Figure 2-1 shows the urinary excretion of uranium after an acute inhalation intake of each material type [ORAUT 2025a]. While the urinary content of Type F material initially drops quickly after the end of intake, it eventually is excreted at a greater rate than Types M and S. Note that the figure depicts uranium urinary excretion; other radionuclides exhibit behavior in accordance with their own biokinetic models.

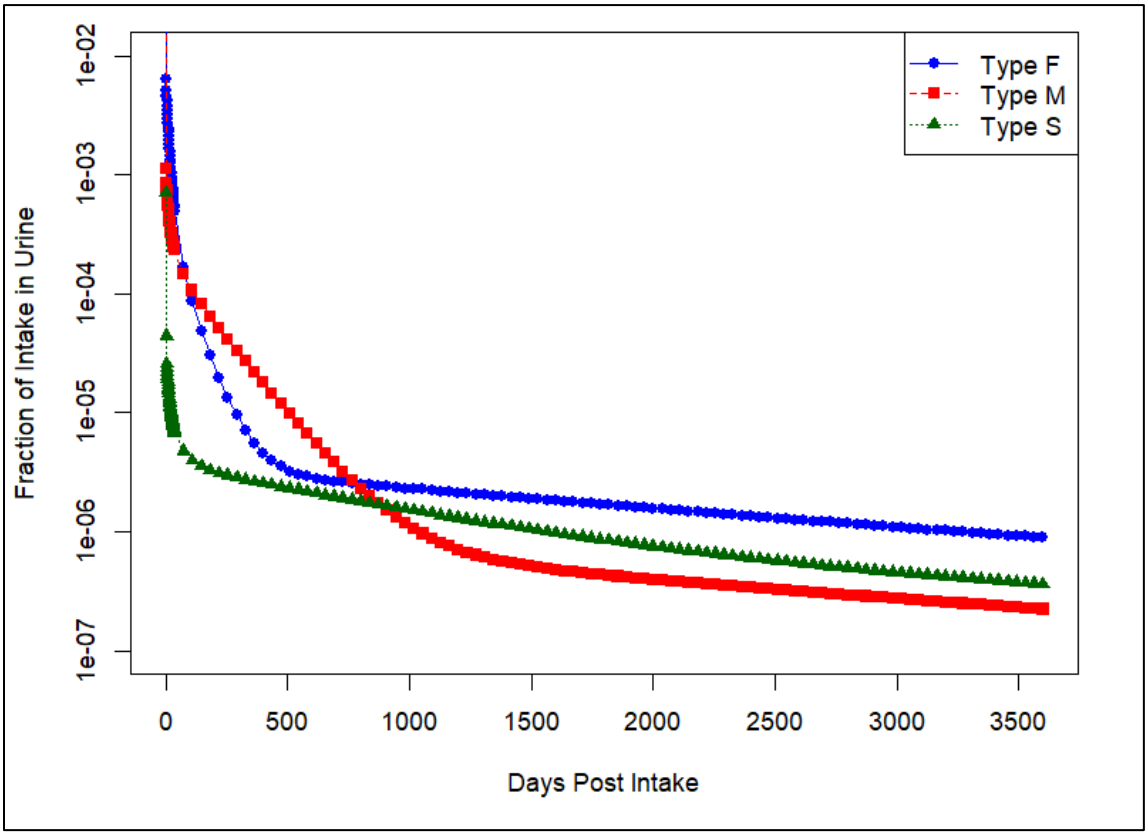


Figure 2-1. Comparison of urinary excretion of Types F, M, and S uranium after an acute intake of unit activity. Attachment E contains an extended description.

2.4 BIOASSAY

2.4.1 In Vitro Bioassay

In vitro bioassay refers to samples that are removed from the body for analysis. The most common sample type is urine and second is feces. Other examples include blood, breath, and tissue samples; these are typically used as indicators of an intake but generally are not used in a dose assessment.

2.4.1.1 Urinalysis

Urine sampling is the most common bioassay method across the DOE complex and was the first bioassay method implemented at most of the covered facilities. Samples might have been collected in the workplace or at home. Translocation of material to the bladder and subsequent elimination via urine occurs over many hours, so urine collected within 24 hours of an acute intake must be used with extreme caution, and those collected within a few hours of intake are often not representative of the intake.

Samples can be in a variety of forms:

- *Spot.* This contains a single voiding of the bladder.
- *24-hour.* This includes all voids collected over a 24-hour period. By convention, the sample date (and sometimes time) of the last void is used to indicate the sample date and is entered in IMBA as the measurement date.
- *Simulated 24-hour.* This varies among sites, but most often consists of samples collected over two consecutive nights. This includes the last void before going to bed, the first void of the morning, and all voids in between. This is intended to replace a true 24-hour sample, which is less convenient for the worker to collect. There are some sites that also use simulated 12-hour samples; these are doubled to determine the 24-hour activity.
- *Minimum volume.* A worker might be given a bottle and asked to fill it rather than being given a specified time. Common requested volumes are 500 mL and 1 L.

While urinalysis is not always the most sensitive bioassay technique, it can detect many nuclides for long periods after an intake and excretion is far more consistent than in feces, so this is often the preferred measurement type for an intake assessment.

2.4.1.2 Fecal Samples

Fecal analysis is a very sensitive bioassay method when samples are collected shortly after an intake because more than half of the material deposited in the upper respiratory tract clears rapidly to the stomach and GI tract. Total fecal elimination for the first few days after an exposure can provide the earliest and most accurate assessment of an intake if all excreta for several days is collected. For this reason, it is an important part of a special bioassay program that is implemented in response to an incident. However, interpretation of a result without a known time of intake, which typically is the case with a routine or confirmatory sample, can yield large uncertainties because fecal voiding patterns are highly variable both among individuals and for a given individual. Additional uncertainty is introduced if only a portion of a sample is analyzed because feces are typically nonhomogenous.

2.4.2 In Vivo Bioassay

In vivo bioassay is the direct measurement of activity from radioactive material that is deposited in the body. Techniques include a whole body count, chest count, or specific organ measurement with the most common being thyroid. In vivo counting is a late entry into bioassay methods, having been developed in the late 1950s, and was not common until the 1960s.

2.4.2.1 Whole Body Counts

A whole body count measures high-energy gamma rays emitted from anywhere in the body, although some counters are collimated so that the view is the torso only. The set point for the lower end of the energy range depends on the nuclides of interest, the detector types, and the site-specific instrumentation settings, but it is often around 100 keV.

2.4.2.2 Chest Counts

Chest counts are intended as a measure of material in the lungs but also include any activity in the ribs and nearby lymph nodes. Low-energy gammas are of interest in these counts, and they are primarily used for the detection of long-term retention of plutonium or uranium in the lungs. Note that Type F material is retained in the lungs for a very short time, so a chest count is not an appropriate technique for monitoring or assessing intakes of Type F material either acute or chronic. Attachment A contains additional discussion and graphs of the retention of all three ICRP material types in the lungs. Type SS is also addressed.

2.4.2.3 Organ Counts

Thyroid counts are frequently used to assess intakes of iodine, which concentrates in the thyroid.

Some sites performed skull counts to determine how much activity was deposited in the bones. This monitoring method is not common and typically is not used in dose reconstruction. Contact the PIDS for guidance if a case involves skull count results.

2.4.2.4 Wound Counts

Some sites might have performed wound counts to help determine how much activity was deposited at a wound site and to aid in the decision for actions to be taken (e.g., cleansing of the area, excision, and chelation). These results are not used directly in a dose reconstruction but can be used to help determine if an intake has occurred and might provide useful information for assessing intakes where chelation was used.

2.4.3 Medical Administration Interference

According to the World Nuclear Association, “Nuclear medicine uses radiation to provide diagnostic information about the functioning of a person's specific organs, or to treat them. Diagnostic procedures using radioisotopes are now routine” [WNA 2023]. Activity administered for medical tests is often significantly larger than quantities found in operational settings, so it can overwhelm an in vivo count or urine sample for a while after the test is conducted. A medical administration performed within a few half-lives before a measurement could invalidate the results because of interference. Mention of a medical procedure is most often seen on an in vivo measurement report where the effect can be seen immediately. The notation can be something like “Tl stress test.” Common nuclides include ^{99m}Tc, ^{201}Tl, and ^{131}I. WNA [2023] includes an extensive list of additional radionuclides that are used in nuclear medicine.

Note that in some cases there are impurities in the administered activity that are small by medical standards but can be easily detected in an operational setting, especially when their half-lives are much longer than the primary radionuclide. For example, the documentation for a medical administration of thallous chloride (^{201}Tl) indicates a ^{202}Tl contaminant of <1.2% [Lantheus 2018]. The ^{202}Tl contaminant has a half-life of 12.23 days, 4 times longer than the 72.9 hours of ^{201}Tl [ICRP 2008]. Further, the intensity of the primary gamma emitted from ^{202}Tl is 9 times that of the primary ^{201}Tl, which makes it easier to detect [ICRP 2008].

Contact the PIDS if interference from a medical administration is possible.

2.5 POSITIVE AND NEGATIVE RESULTS

2.5.1 Definitions

2.5.1.1 Censored Data

The numerical value of a censored datum is only partially known. For example, a measurement can be reported as

1. <10 (left censored),
2. >10 (right censored), or
3. <10 and >5 (interval censored).

In the first two, 10 is the censoring level. In the third, 5 and 10 are the censoring levels. In practice, the right-censored example (2) is rarely seen and is not useful for assessing a dose other than to indicate that an intake likely occurred.

2.5.1.2 Decision Level

Decision level (DL) is defined in ANSI/HPS N13.30-2011, *An American National Standard – Performance Criteria for Radiobioassay*, as the number of counts measured or final instrument measurement of a quantity of analyte at or above which a decision is made that the analyte is definitely present [Health Physics Society (HPS) 2011]. The term “definitely present” in the standard is somewhat ambiguous, but what is meant is that the analyte is deemed to be present with a given probability α of being wrong. The probability α is often called the false-positive rate, and it describes the long-run frequency of deciding the analyte is present in a sample when in reality none is present. In an operational setting, the result of a specific analysis is compared to the DL to decide if the analyte is present in that sample.

2.5.1.3 Minimum Detectable Amount

MDA is defined in ANSI/HPS N13.30-2011 as being the smallest amount (activity or mass) of an analyte in a sample that will be detected with a probability β of nondetection (Type II error) while accepting a probability α of erroneously deciding that a positive (nonzero) quantity of analyte is present in an appropriate blank sample (Type I error) [HPS 2011]. In other words, the MDA is the amount of analyte in samples that would produce results below the DL a fraction β of the time over the long run. The MDA is used to characterize the detection capabilities of the process, and HPS [2011] states that it should not be compared to a specific analytical result to make the detection decision. However, it is appropriate for the determination of missed dose, the purpose of which is to account for the detection capabilities and limitations of the bioassay monitoring that was performed.

2.5.1.4 Reporting Level

For this Project, a reporting level is defined as the level below which a value is not recorded for a bioassay result in the worker's record; that is, the result is censored at the reporting level. In some instances, a site might have applied a reporting level that is greater than the MDA. In such cases, only measurements with values that exceed the reporting level are recorded in the worker files. That is, results less than the reporting level are considered less-than values and are recorded as zero or "<" the reporting level. In these cases, the reporting level, not the MDA, is used to determine the missed dose. This practice of censoring the data is most common when the nuclide is easily detected, such as ^{3}H, and a result at the MDA produces a very small dose.

2.5.2 Determination of a Positive Result for Dose Reconstruction

When discussing bioassay analyses for dose reconstruction, a "positive result" typically refers to one that is greater than the MDA or reporting level while a "negative result" is less than or equal to that value. It is possible for a result to be a true negative value (i.e., less than zero). Both uses of these words can be found in Project documents, so it is important to be aware of which meaning is being discussed. Results between the DL and MDA are accounted for in the missed dose calculation, so they are not considered to be positive when performing a dose reconstruction.

Many sites recorded the value of the MDA without a preceding "<" in the worker records for activity measured at less than the MDA. That is, the results are censored at the MDA. In such cases, a result recorded with a value of the MDA is treated as a negative result. If it is clear that this is *not* the case, a result equal to the MDA should be treated as positive. If it is unclear, a result equal to the MDA is considered negative.

Sample-specific information takes precedence over the default MDA in the site profile. Note that in some cases the MDA can be calculated from other sample-specific information. See Section 3.4.5 for guidance.

In the early years of the weapons complex, calculations were not standardized and most sites used the equivalent of the MDA to determine if activity had been detected. In more recent years, sites have used the DL to make the decision about the presence of activity. As noted above, the MDA is used for dose reconstruction purposes.

2.5.3 Occupational Uranium Intakes

Uranium is naturally occurring in the earth's crust and can be found in the urine and feces of people with no occupational exposure. This uranium background can vary greatly depending on local geology, eating habits, and water source. ICRP Publication 23 indicates that published values show a range of urinary excretion spanning more than 3 orders of magnitude and provides discussion on various sources of intake [ICRP 1975]. Daily losses via fecal excretion are of a larger magnitude but cover a smaller range of values.

Sites are only required to track and report on occupational exposures to radioactive materials. Therefore, sites might be interested in determining this uranium background and might have set levels below which they considered a result attributable to background. However, because of the large variability between individuals, and to be favorable to the claimant for dose reconstruction purposes, any result other than a baseline sample that is positive (as defined in Section 2.5.2) is assumed to be due to an occupational exposure.

2.5.4 Nonoccupational Cesium Intakes

Fallout affected everyone in North America, and body burdens of ^{137}Cs measurable in whole body counters were common in the 1960s and 1970s. Table 2-1 provides mean body burdens of ^{137}Cs for the United States from National Council on Radiation Protection and Measurements (NCRP) Report 94 for the years when this fallout was most likely to produce interference with occupational whole body count results [NCRP 1987]. If whole body count results show detection of only ^{40}K and ^{137}Cs, and the ^{137}Cs result is less than the value in Table 2-1, the ^{137}Cs result can be assumed to be due to fallout. Values exceeding these levels are assumed to be due to occupational intakes, and the full value is used in the assessment (i.e., the values in the table are not subtracted from the measured result). If other fission or activation product radionuclides are present in the whole body count, assume any positive ^{137}Cs result is from occupational sources and assess based on the full value.

Table 2-1. Mean body burden of ^{137}Cs in the United States.[a]

Year	Body burden (Bq)	Body burden (nCi)
1953	10	0.27
1954	40	1.08
1955	80	2.16
1956	160	4.32
1957	190	5.13
1958	240	6.48
1959	300	8.1
1960	250	6.75
1961	170	4.59
1962	220	5.94
1963	400	10.8
1964	700	18.9
1965	590	15.9
1966	360	9.72
1967	210	5.67
1968	130	3.51
1969	100	2.7
1970	100	2.7
1971	100	2.7
1972	100	2.7
1973	100	2.7
1974	60	1.62
1975	40	1.08
1976	60	1.62
1977	40	1.08

a. Source: NCRP Report 94 [NCRP 1987, Table B5].

2.6 TYPES OF DOSES AND ASSOCIATED DISTRIBUTIONS

This section describes the different types of doses that can be assigned to a worker, which depend on several factors including the site of employment, available data, and potential for exposure. Appropriate distributions to be entered into the Interactive RadioEpidemiological Program (IREP) are discussed.

Sections 5.0 and 6.0 discuss the applicable conditions for assigning each type of dose and provide guidance on the calculations.

2.6.1 Philosophy of Internal Dose Assignment

The general assumption for internal dose reconstruction is that a worker was constantly exposed to at least some level of airborne radioactivity throughout employment. At a minimum, ambient background levels are assigned. This might not be an accurate representation of the exposure pattern, but it best approximates the intakes given the available information and provides for consistency across all individuals.

2.6.2 Unmonitored Dose

Unmonitored dose is the potential dose that could have been received by a worker but for which no monitoring of the individual was performed or monitoring data are not available but for whom doses can be reconstructed using available methods.

2.6.3 Missed Dose

Missed dose is based on the potential intake that could have been received by a bioassay program participant but, because of limitations in the monitoring system or a policy decision to use a reporting level, was undetected (i.e., activity went undetected because it was less than the detection limit or unreported because it was less than the reporting level).

In an operational setting, the application of dosimetric and biokinetic models to the MDA of a bioassay technique is referred to as the minimum detectable dose (MDD). The MDD is used for selecting bioassay programs to meet given monitoring requirements and should not be assigned to workers. However, a similar quantity is included in the dose reconstruction assessment and assigned as the missed dose to the worker as an assumption that is favorable to the claimant. Note that if a fitted dose has been calculated for all periods of potential exposure, there is no missed dose.

Missed dose is assigned using actual worker in vivo or in vitro bioassay measurements that are *negative* (i.e., less than or equal to the MDA or reporting level) and worker-specific employment information. The “true” dose is assumed to fall between zero and that which would result in the prediction of bioassay results equal to the MDA. In many instances, data below the MDA were censored and only the MDA value was reported. Therefore, to standardize the missed dose calculation, one-half the censoring value is used (e.g., half of the MDA or half of the reporting level). A value of one-half the MDA is substituted for the result and used to calculate the missed dose. A triangular distribution is then assigned to the dose to account for uncertainty in the bioassay result. The triangular distribution is assigned in IREP with a minimum of zero, a maximum based on results at the MDA, and a mode based on results at one-half the MDA.

Section 5.2 provides information on the calculation of missed dose.

2.6.4 Fitted Dose

For this Project, “fitted dose” is the term for doses that are calculated from positive bioassay results (although some <MDA results might also be used in the fitting of the intake), which typically confirms that an intake did occur. This is calculated *independently* of the missed dose and typically is entered as a lognormal distribution in IREP. A geometric standard deviation (GSD) of 3 is assigned to account for the uncertainty in the bioassay measurements and the biokinetic models [ORAUT 2024].

Section 6.0 provides guidance on fitting positive bioassay results.

2.6.5 Environmental Intakes

Internal radiation doses to some workers were limited to doses from inhalation of airborne radionuclides in the ambient environment from site operations or contamination rather than from localized airborne radionuclides from uncontained radioactive materials in the workplace. For these workers, assignment of environmental dose only is appropriate. Environmental dose is also assigned to monitored individuals if there are nuclides in the environmental suite of radionuclides that have not been assigned through some other method in the dose reconstruction.

Environmental intakes are documented in the site profile. Direction for assigning these intakes is given in the site profile.

2.6.6 Co-Exposure Dose

Co-exposure intake distributions are developed from available dosimetric data from DOE or AWE sites. DCAS-IG-006, *Criteria for the Evaluation and Use of Co-Exposure Datasets* [NIOSH 2020], provides guidance for the evaluation of personnel monitoring data to be used in the reconstruction of doses to unmonitored workers. Additional documents specify the statistical techniques to be used in the development of these models. Site-specific TIBs and site profiles are available for some sites and might provide tables of intakes and associated GSDs based on co-exposure dosimetry analyses.

Workers with a significant potential for intake are typically assigned doses at the 95th percentile with a constant distribution, while those with less potential are assigned the 50th percentile with a lognormal distribution. The site profile might provide more detailed guidance.

3.0 INTAKE ASSESSMENT PARAMETERS

This section provides information necessary for calculating an intake from bioassay results. The intake activity is subsequently used for calculating the dose. When values are unknown, default values specified here or in the site profile should be applied. Specific knowledge of an individual claim or site takes precedence over default values (see Section 1.2).

3.1 INTAKE MODE

The five intake routes described in IMBA are used for convenience:

- Inhalation is the most common route for workplace intakes. Absorption type, discussed in Section 3.3, is used to describe the movement of material from the lungs to the blood.

- Ingestion can be associated with an inhalation intake. The fractional uptake of a radionuclide from the SI region of the GI tract to blood is described by the parameter f1. When an inhalation intake is assigned based on air monitoring rather than bioassay data, an additional ingestion component might need to be assigned. See OCAS-TIB-009, *Estimation of Ingestion Intakes,* for guidance [NIOSH 2004].

- Injection is the entry of material directly into the bloodstream; this is sometimes referred to as absorption. Part 1 of ICRP Publication 56 treats the inhalation of tritium oxide as injection [ICRP 1989].

- Wound is used to model the retention of material at a wound site, where there is a delay of entry into the intravenous blood circulation.

- Vapor is a specific instance of an inhalation intake rather than a separate route; it is defined as the gaseous form of substances that are normally in liquid or solid form. Iodine is typically modeled using this intake route. Elemental iodine is then selected as the material type using the ICRP Defs Load button in IMBA.

In the absence of information about how an intake might have occurred, inhalation is the default assumption when starting with bioassay data because this is the most likely route of entry in an occupational setting.

3.2 PARTICLE SIZE DISTRIBUTION

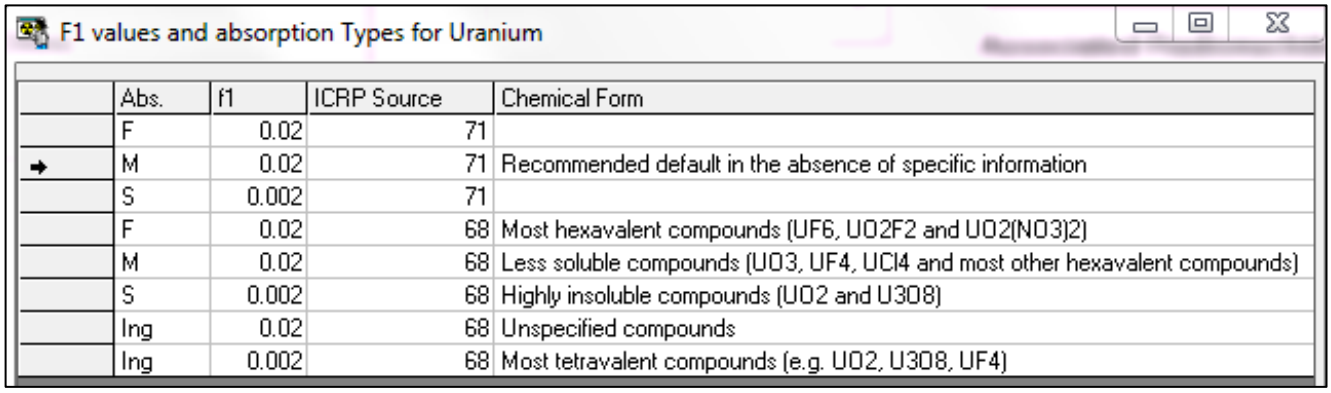
The particle size distribution dictates the assumed deposition pattern of inhaled material in the various regions of the respiratory tract. For occupational exposures, the ICRP Publication 66 default value is a 5-µm activity median aerodynamic diameter (AMAD) [ICRP 1994a]. This value should be used for evaluating inhalation intakes in the absence of known information as documented in the site profiles or the claim file. Note that previous ICRP guidance was based on a default of 1-µm AMAD [ICRP 1979], so most site evaluations use this as a default during the relevant timeframe. There might be special cases where a site performed a particle size study for a specific location or incident; these are documented in the site profile or claim file. Where the site used a default value, the current default value of 5 µm is applied to the dose reconstruction.

3.3 ABSORPTION TYPE

3.3.1 Standard ICRP Types

As discussed in Section 2.3, ICRP Publication 66 describes three types: F (fast solubilization), M (moderate solubilization), and S (slow solubilization) [ICRP 1994a]. The assignment of an element to one or more of these categories is based on the chemical form of the material. The recommendations of ICRP Publication 68 [ICRP 1994b] are used for this Project because they address worker intakes rather than those of members of the public. These recommendations are in IMBA. Figure 3-1 shows an example from the IMBA ICRP Defaults selection menu. If the chemical forms of material at a particular site are known, they are documented in the site profile and should be applied as specified in that document.

For the majority of claims, it is likely that the material type is unknown or that the individual worked in multiple areas, which makes exposure to multiple types possible. In such instances, an assessment of each type to which the element is assigned in ICRP Publication 68 [ICRP 1994b] should be made, and the type that results in the largest dose to the organ of interest should be selected. Note that there are additional considerations when multiple cancers are contributing to the probability of causation (POC); see Section 7.3.2 for guidance. See Attachment B for a complete list of Publication 68 material types for all elements of interest. The exception to the use of Publication 68 types is in the case of a radionuclide where trace atoms are tightly bound in a matrix of another nuclide. The primary example of this is ^{241}Am in a plutonium matrix. Americium is only assigned to Type M by ICRP Publication 68, but evidence has shown that when bound in a Type S plutonium matrix, americium also exhibits Type S behavior. Therefore, ICRP Publication 71 [ICRP 1995] Type S is selected for ^{241}Am when in a Type S plutonium matrix.



F1 values and absorption Types for Uranium

	Abs.	f1	ICRP Source	Chemical Form
	F	0.02	71	
→	M	0.02	71	Recommended default in the absence of specific information
	S	0.002	71	
	F	0.02	68	Most hexavalent compounds (UF6, UO2F2 and UO2(NO3)2)
	M	0.02	68	Less soluble compounds (UO3, UF4, UCl4 and most other hexavalent compounds)
	S	0.002	68	Highly insoluble compounds (UO2 and U3O8)
	Ing	0.02	68	Unspecified compounds
	Ing	0.002	68	Most tetravalent compounds (e.g. UO2, U3O8, UF4)

Figure 3-1. IMBA window showing uranium default absorption types and f1 values. Attachment E contains an extended description.

3.3.2 Additional Types

3.3.2.1 Plutonium

The behavior of some forms of plutonium in the respiratory tract are not adequately described by the standard ICRP models, so alternative models have been developed.

When assessing ^{239}Pu, Type SS could need to be considered in addition to Types M and S. This type potentially applies to all methods of assigning intakes including co-exposure doses and efficiency methods. ORAUT-OTIB-0049, *Estimating Doses for Plutonium Strongly Retained in the Lung* [ORAUT 2020a], discusses conditions under which this type applies and appropriate assessment methods. Associated ^{241}Am and other plutonium isotopes are also assessed as Type SS when part of the ^{239}Pu mixture. This model cannot be run in IMBA; IDOT is used for intake assessments [NIOSH 2019].

Plutonium-238 also has alternate material types that are applied at specific sites under certain conditions. Details of these models and their applicability are documented in DCAS-RPT-005, *Alternative Dissolution Models for Insoluble Pu-238* [NIOSH 2018a]. These models can be run in IMBA and, for convenience, files containing the appropriate parameters have been created. The models are identified as “Mound” and “LANL” in the report. These correspond to MHI (Mound highly insoluble) and NM (nonmonotonic), respectively, in the chronic annual dose tool Web CAD and the IMBA model files. IMBA input files containing these models are in *IMBA Alternative Nuclide Models H-3 and Pu-238* [ORAUT 2025b].

3.3.2.2 Uranium Aluminide

Uranium aluminide (UAl_x) has been found to be less soluble than other forms of uranium. A modification to the ICRP Publication 66 lung model to specifically address the biokinetics of UAl_x was developed by Leggett et al. [2005]. Details of this model, referred to as Type K, and its application are described in the internal dose technical basis document for Area IV of the Santa Susana Field Laboratory [ORAUT 2010].

3.3.2.3 Special Tritium Compounds

Stable metal tritides (SMTs) and organically bound tritium (OBT) are special forms of tritium compounds. Modelling of SMTs is addressed in ORAUT-OTIB-0066, *Calculation of Dose from Intakes of Special Tritium Compounds* [ORAUT 2020b]. IMBA input files containing these models are in ORAUT [2025b].

Intakes of OBT can be modeled directly in IMBA using the following steps:

1. Select **Hydrogen-3 (organic)** as the Indicator Nuclide.
2. Select **Vapor** as the Route.
3. Load the ICRP defaults by clicking the icon at the top of the screen.
4. Select **Organic compounds** from the menu that pops up (this is the only option).

3.3.3 Intakes of Mixtures

3.3.3.1 General

When a mixture of nuclides is bound in a single matrix, the contaminants typically are assumed to be relatively tightly bound but not to behave outside of their own models. Therefore, the same material type should be applied to the contaminants as that selected for the host radionuclide. If ICRP Publication 68 does not assign the contaminant nuclide to the chosen host material type [ICRP 1995], the closest absorption type should be selected. An example of these mixtures is recycled uranium (RU). (See Section 9.2.1.2 for additional information on RU.) Table 3-1 provides guidance on material types for several elements that might be included as RU contaminants. Uranium is the host material in this instance, and can exist as Type F, M, or S according to ICRP Publication 68. Plutonium is only assigned to Types M and S by the ICRP in Publication 68, so when it is a contaminant in a Type F uranium mixture, plutonium should be assigned to Type M because that is the closest absorption form to Type F. Neptunium and americium are only assigned to Type M in ICRP Publication 68, so Type M should be selected for all RU mixtures regardless of uranium form. Other contaminant nuclides should follow the same rule. See Section 9.4 for additional discussion on mixtures. Contact the PIDS for assistance with other nuclides or mixtures.

Note that ^{239}Pu and thorium mixtures are an exception, as discussed in Sections 3.3.2.1 and 3.3.3.2, respectively.

Table 3-1. Selected absorption types for RU contaminants.

Uranium type	Contaminant and absorption type	
F	Pu	M
	Np	M
	Tc	F
	Th	M
	Am	M
M	Pu	M
	Np	M
	Tc	M
	Th	M
	Am	M
S	Pu	S
	Np	M
	Tc	M
	Th	S
	Am	M

3.3.3.2 Thorium

Thorium progeny that grow in *before* the thorium aerosol material is deposited in the lungs (i.e., are present in the inhaled air) are assumed to be separate materials. Once inhaled, each of these radionuclides is assumed to have its own absorption type that can be different than that of the parent thorium.

Thorium progeny that grow in *after* the thorium aerosol is deposited in the lungs (i.e., are born in the lungs) are assumed to be locked in the matrix of the aerosol and have the same absorption type as the parent thorium [ORAUT 2023].

3.3.3.3 Wounds

IMBA includes the NCRP default wound models. The same selection should be made for all nuclides in a mixture. Note that these are not included in Web CAD, so IMBA must be used for the dose assessment. Section 10.1 contains discussion on wound assessments.

3.3.4 Ingestion Intakes

When ingestion intakes are assigned along with an inhalation mixture, the selected f1 values are matched to the f1 values assigned to the absorption types. Figure 3-1 in Section 3.3.1 includes the uranium f1 values. Table 3-2 contains a summary of selected f1 values for an RU mixture.

Table 3-2. Selected f1 values for RU contaminants.

Uranium type	Corresponding f1 value	Contaminant and f1 value	
F, M	0.02	Pu	0.0005
		Np	0.0005
		Tc	0.8
		Th	0.0005
		Am	0.0005
S	0.002	Pu	0.00001
		Np	0.0005
		Tc	0.8
		Th	0.0002
		Am	0.0005

3.4 BIOASSAY DATA

3.4.1 Invalid Data

The following entities were used by EEOICPA covered facilities as contractors or vendors to analyze various types of samples. These vendors were found to have produced potentially falsified or otherwise compromised results:

- Controls for Environmental Pollution (CEP), Santa Fe, New Mexico; operated from the mid-1970s to 1994.
- Alvin E. Blackwell (AEB) Consultants, Little Rock, Arkansas; operated from 1990 to 1993.

Therefore, no bioassay results reported by CEP or AEB for any site or timeframe are to be used for a dose assessment [NIOSH 2025].

Note that while the above results cannot be used to assess an intake, the presence of bioassay can be an indication of the potential for intakes. Refer to NIOSH [2025] for additional information about results from these laboratories.

3.4.2 Offsite Data

Bioassay measurements do not have to be performed on site to be usable in a dose reconstruction. They do need to be from a reputable laboratory and able to be interpreted. Contact the PIDS for assistance if offsite measurements are encountered and not addressed in the site profile.

3.4.3 Positive Baseline Samples

A site might collect an in vitro sample or perform an in vivo measurement before a worker begins work where there is a potential for intakes at that site. These typically are used to determine if there are intakes from previous employment or a nonoccupational source that might interfere with future measurements.

A positive baseline sample from a worker with no previous employment at a covered facility has no impact on the intake assessment because, by definition, it precedes the potential for exposure. If the next sample (i.e., taken after work began) for the same radionuclide is also positive, contact the PIDS for assistance.

If a positive baseline is encountered from a worker who was previously employed at a different covered facility, contact the PIDS for assistance.

Some sites performed prejob sampling, which might precede any potential for intake at the site or could be taken when the worker moved to a new location within the site. This section does not apply if the worker had a previous potential for intake at the given site.

3.4.4 Postemployment Data

Bioassay samples do not need to have been taken during the exposure or employment period to be useful for dose reconstruction. Some sites have a former worker medical surveillance program with continued monitoring after employment; these are addressed in the respective site profiles. A worker might also have later bioassay from employment at a noncovered facility.

Samples collected after employment must be reviewed to ensure there was no potential for exposure at a facility outside of the EEOICPA program after the covered employment period. Contact the PIDS if assistance is needed.

3.4.5 Calculation of Minimum Detectable Amount

If an MDA is not listed for a measurement, it might be possible to determine it from other information [ORAUT 2022]. Note that this direction applies in general to bioassay results beginning in 1992. Error calculations are more likely to be standardized in this era, which allows for consistency among the sites. Before 1992, this should be applied on a site-by-site basis as documented in the site profile or dose reconstructor guidance documentation.

If a DL is recorded for a sample, the MDA is calculated as:

$$MDA = 2 \times DL \tag{3-1}$$

If only the standard deviation (commonly listed as the error associated with a result) for the measurement is provided (i.e., neither an MDA nor DL are given), the MDA is calculated as:

$$MDA = 3.29 \times \sigma \tag{3-2}$$

where

σ is standard deviation

Note that site reports often list the 2σ error, so the value is divided by 2 before multiplying by 3.29 when that occurs. See the site profile or dose reconstruction guidance for information on what error the site reported.

When a sample is counted multiple times, determination of the MDA using the error is more complex than shown above. Attachment C contains a more detailed discussion of the issue. Contact the PIDS for guidance on determining if a sample with multiple counts is positive.

3.4.6 Use of Chest Counts for Intake Analysis

Material type is associated with the rate at which material is removed from the lungs. As noted in Section 2.4.2.2, Type F clears very rapidly: in just a few hours very little of the intake remains in the lungs (see Figure A-1). This is independent of the radionuclide; the lung model is the same for all. Because it leaves the lungs so rapidly, a chest count cannot be used to assess a Type F intake. The activity will have moved to other parts of the body in the time it would take for a person to get to the counter. When there is a potential for intake and there are both urine samples and chest counts, only the urine samples are used for the Type F assessment. If there are no urine samples, the worker is

considered unmonitored for Type F material; co-exposure data, or some other estimate of unmonitored intake, should be used if there is a potential for an intake of Type F material.

Type M clears the lungs more slowly, so a chest count can be used for assessing an intake. Because the material clears substantially in the first year, the rules for short-lived or short-retained radionuclides apply. For example, gaps of greater than 2 years between results are considered to be unmonitored. Section 5.4 contains additional guidance on this.

Types S and SS are retained in the lungs for very long periods after an intake, so chest counts can be used to assess intakes many years after an intake.

3.4.7 Normalization of In Vitro Bioassay Results

IMBA requires urine or fecal sample results to be entered as activity excreted over a 24-hour period. If results are not reported per day (or 24 hours), they must be normalized before entering into IMBA. When determining if a result is positive (i.e., greater than the MDA), like units must be applied. For example, if the result is in units of dpm/L, it must be compared to an MDA in units of dpm/L.

Note that the adjustments described below do not apply to a sample collected within the 24 hours after an acute intake because the excreted fraction is rapidly changing and is therefore not linear. Contact the PIDS for assistance if such a result is needed for the assessment.

3.4.7.1 Urine Results Reported "Per 24 Hours" or "Per Day"

If sample results are reported "per 24 hours" or "per day," no adjustment is performed regardless of the reported volume.

3.4.7.2 Urine Results Reported Per Volume

3.4.7.2.1 Known 24-Hour Collection

If the results are reported per volume (e.g., per liter) and the sample collection period is a known 24-hours, then use the reported sample volume for the per day normalization.

Example 1:

Urine result is 1 dpm/L and reported sample volume is 1.8 L.

Normalized result is $1\ \text{dpm/L} \times 1.8\ \text{L/d} = 1.8\ \text{dpm/d}$.

Example 2:

Urine result is 0.01 dpm/mL and reported sample volume is 1,200 mL.

Normalized result is $0.01\ \text{dpm/mL} \times 1{,}200\ \text{mL/d} = 12\ \text{dpm/d}$.

3.4.7.2.2 Unknown Collection Period

If a sample volume is reported but there is no indication of the collection period and the site profile and dose reconstructor guidance document contain no additional information, default values must be used.

Title 42 *Code of Federal Regulations* (C.F.R.) Part 82 requires the use of ICRP models for calculating internal doses and specifies the use of certain publications [42 C.F.R. 82, 2019]. The biokinetic models and subsequent dose coefficients in those publications are based on male physiology and

anatomy. Therefore, when urinalysis results need to be adjusted and no other information is available, the results should be normalized to 24 hours using a conversion factor of 1.4 L/d for all individuals, male or female.

Normalize the result when the sample volume is less than those listed above. If the sample volume is greater than above, assume a 24-hour collection period and do not normalize the values. This is favorable to the claimant because normalization would reduce the reported value.

Example 3:

Urine result is 1 dpm/L and reported sample volume is 1.8 L.

Volume is >1.4 L/d, so the normalized result is $1\ \text{dpm/L} \times 1.8\text{L/d} = 1.8\ \text{dpm/d}$.

Example 4:

Urine result is 1 dpm/L and reported sample volume is 1.2 L.

Volume is <1.4 L/d, so the normalized result is $1\ \text{dpm/L} \times 1.4\ \text{L/d} = 1.4\ \text{dpm/d}$.

3.4.7.3 Urine Results Reported "Per Sample"

Results reported "per sample" with an unknown collection period are normalized in the same manner as those reported per volume: Normalize the result when the sample volume is less than 1.4 L and do not adjust the result if the sample volume is greater than 1.4 L.

Example 5:

Urine result is 1 dpm/sample and reported sample volume is 1.8 L.

Volume is >1.4 L/d, so no normalization is needed. The result is assumed to be 1 dpm/d.

Example 6:

Urine result is 1 dpm/sample and reported sample volume is 1.2 L.

Volume is <1.4 L/d, so the normalized result is $(1\ \text{dpm}/1.2\ \text{L}) \times 1.4\ \text{L/d} = 1.17\text{dpm/d}$.

3.4.7.4 Fecal Results

As indicated in Sections 2.4.1.2 and 5.2.2, interpretation of fecal results can be challenging because fecal voiding patterns are highly variable both among individuals and for a given individual. Efforts should be made to determine the collection period. Contact the PIDS if assistance is needed.

If normalization of a fecal result is deemed necessary, it would be performed as above for urine samples using a default mass of 135 g/d.

3.5 SELECTION OF APPROPRIATE ORGAN FOR DOSE ESTIMATE

As noted in Section 2.2, ORAUT-OTIB-0005 [ORAUT 2019] provides direction on the appropriate organs and tissues to be modeled. When the highest nonmetabolic organ is specified, dose reconstructors do not need to make this determination because Web CAD selects the nonmetabolic organ with the largest dose for the selected ICD-10 code when appropriate.

4.0 INTAKE AND DOSE ASSESSMENT SOFTWARE

This section describes the uses and limitations of the primary intake and dose assessment tools for dose reconstruction.

4.1 INTEGRATED MODULES FOR BIOASSAY ANALYSIS

IMBA is the primary tool for assessing intakes and creating tables of dose conversion factors to be used in Web CAD. It can also be used directly for assessing doses. Additional information about IMBA and how it functions are in the user manual [ACJ 2005].

4.1.1 Data Type

IMBA defines the following data types:

- *Real.* This type indicates a valid result to be used in the intake calculation. IMBA uses this type of result as is. There must be at least one Real result entered for IMBA to perform an intake calculation.

- *<LOD.* This stands for “less than the limit of detection,” and the Project uses this type for results less than or equal to the MDA (or reporting level, as appropriate; see Section 2.5.1.4) as described in Section 6.0 on fitting positive bioassay data. The sample MDA is entered for the Measurement Rate or Measurement Value for results marked as this Data Type. IMBA uses a maximum likelihood method for estimating intakes; <LOD results are treated as a distribution between zero and the LOD in the intake calculation.

- *Excluded.* IMBA ignores all results with this Data Type when calculating the intake but displays them in the graphs. This type is most often used for overestimates to ensure results are overpredicted by the projected values.

4.1.2 Measurement Error

The inverse square of the Measurement Error is the weighting factor applied to the Measurement Rate for the intake fit. The larger the Measurement Error, the smaller the relative weight given to the result in the intake calculation. In other words, IMBA attempts to fit the results with the smallest errors first, while those with larger errors have less emphasis placed on them. The fit to the data tends to move toward the results with the smallest errors on an absolute scale. The Measurement Error cannot be zero: this causes IMBA to display an error message.

This field can be entered manually by typing a value in the box. IMBA also allows the user to generate the Measurement Error by selecting the type of error relative to the measured result and a factor *K* by which it is multiplied. Options for generating the Measurement Error are:

- *Uniform Absolute.* The selected value of *K* is entered as the Measurement Error for all selected results. This weights all values equally.

- *Uniform Relative.* The value entered in the Measurement column is multiplied by *K* to obtain the Measurement Error.

- *Square Root.* The square root of the Measurement is multiplied by *K* to obtain the Measurement Error.

Click **OK** after selecting the error button and entering the *K* value. Uncheck the "Apply to all" box if different relative errors are desired (e.g., when there is a mix of Real and <LOD results).

For positive (Real) results, the 1-sigma error reported with the result can be used when available. If there are no reported errors with the results, Measurement Error should be calculated using the Uniform Relative option with *K* = 0.3 for all positive (>MDA) results *as a starting point*. Note that in the case where all results are positive (i.e., all are labeled as Real), the value of *K* as applied by the Uniform Relative option is arbitrary; the same intake is calculated from any factor given the same *K* for all results.

Measurement values entered as <LOD should have an error equal to the LOD. This is equivalent to using *K* = 1 with the Uniform Relative option to enter the errors.

Note also that, when only a single result is used for an intake determination, the value for the Measurement Error is not included in the calculation (there is nothing to be weighted) so the intake is the same regardless of the entered error.

A Uniform Relative error with equal values of *K* for all results might not be a reasonable estimate if there are well-defined peaks or if results vary by more than an order of magnitude. For example, given two results with values of 100 and 10, the fit would go closer to the 10 if a Uniform Relative error with *K* = 0.3 was applied because these would be treated as 100 ±30 and 10 ±3. Larger values have more precise statistics and might need to be assigned relatively smaller errors to obtain the desired fit (e.g., overestimate, underestimate, best estimate). For a best estimate, alternative values for the error can be tried if it is not consistent with the results (e.g., the majority of results appear to be underpredicted or the larger results are greatly underpredicted). Application of a 10% error to the largest results while retaining a 30% error on the smaller positive results might improve the fit. Other values can be tried if this does not provide the desired fit. Use of the Uniform Absolute option, with the same value entered for all results, yields an unweighted fit (i.e., all results are weighted equally) and might be the best option if all results are Real. Contact the PIDS for assistance if a reasonable fit cannot be obtained.

Note that changing the weighting factor (i.e., the Measurement Error) on the results while keeping all other parameters the same (e.g., intake dates) simply moves the data fit up and down; it does not change the shape of the curve.

4.1.3 Error Distribution

"Normal" is typically selected for the distribution. The Joint Committee for Guides in Metrology (JCGM) states that a normal probability distribution should be used with the point estimate and standard deviation to calculate statistics like the 95% confidence interval (e.g., the 2-sigma interval) for the result [JCGM 2023].

4.1.4 IMBA Limitations

IMBA is known to calculate inaccurate organ doses for some radionuclide and organ combinations (Table 4-1). There are two categories of problems: shared versus independent kinetics and very short-lived radionuclides. In the former case, the current ICRP system models the progeny of a radionuclide using the kinetics of the progeny but, because of limitations in the software, IMBA employs the older model technique of shared kinetics, where the progeny is assumed to follow the behavior of the parent, which results in inaccuracies in some radionuclides with radioactive progeny [ACJ 2004, Appendix C]. In the case of very short-lived nuclides, the reason for the problem is unknown but IMBA incorrectly calculates the annual doses from chronic intakes. Additional details on the limitations are in *IMBA Dose Assessment Limitations* [ORAUT 2009].

Table 4-1. Nuclides for which IMBA calculates incorrect organ doses.

Nuclide	Nuclide	Nuclide
Ac-228[a]	Te-131	Th-232
Pb-210	Te-131m	Th-234
Pr-147[a]	Te-132	U-232
Ra-223	Te-133	U-233[b]
Ra-224	Te-134	U-239[a]
Ra-226	Th-228	
Ra-228	Th-229	

a. Doses due to chronic intakes only; acute intakes are not affected.
b. Type S only.

For these nuclides, annual dose coefficients have been calculated with the DCAL program [Eckerman et al. 2006]. These coefficients have been incorporated into Web CAD, and verification of the values is documented in ORAUT-OTIB-0028, *Validation of DCAL Annual Dose Coefficients* [ORAUT 2008]. The IMBA *intake* calculations are correct, so bioassay results can be input to IMBA for determining the intake.

For the radionuclides in Table 4-1, Web CAD must be used to calculate the dose. IMBA *cannot* be used as an over- or underestimate for these nuclides because there is no consistency in the direction in which IMBA is incorrect (i.e., it is not always high or low for all organs and years for a given radionuclide).

Contact the PIDS when a best estimate is needed for timeframes that cannot be calculated with Web CAD (see Section 4.3 for details on Web CAD limitations). When sending a claim to the PIDS for a best estimate, the following information must be included:

- Nuclide,
- Intake date(s),
- Material type,
- Intake (acute) or intake rate (chronic) as applicable and including units,
- Organ of interest,
- Date of diagnosis, and
- Claim due date.

4.2 INTERNAL DOSIMETRY TOOL

As noted in Section 3.3.2.1, the model that was developed for assessing intakes of strongly retained ^{239}Pu (Type SS) cannot be adapted to IMBA, so IDOT was developed [NIOSH 2019]. It was designed to function in a manner similar to IMBA; input fields have the same names and the general guidance for using IMBA is applicable.

Refer to the IDOT user guide [ORAUT 2020c] for information on the installation and use of the tool.

4.3 WEB CAD

IMBA is designed for an operational radiation protection program, so it is somewhat inefficient for assessing the complete exposure history of a worker. Web CAD was developed as a tool to assist the dose reconstructor in assessing doses and compiling them for input to IREP. It does not perform intake assessments.

Rather than using models to calculate doses, Web CAD uses lookup tables from IMBA or other internal dosimetry software to generate annual organ doses from intakes that the user inputs. This results in some limitations on its use. The tool is based on whole-year intakes and prorates those doses based on fraction of year when intake does not start or occur on January 1 or when the date of diagnosis is not December 31. However, dose is not delivered linearly for a few years after an intake because the material is being transported to various organs. For example, the dose imparted to a particular organ in the first 6 months after an intake might be less than that from the last 6 months, so using half the annual dose for an intake on July 1 (or a chronic intake beginning on July 1) would yield an overestimate in the year of intake. Consequently, its use for partial years results in an approximation of the annual dose that can be an under- or overestimate of that obtained from an exact solution.

4.3.1 Partial-Year Chronic Intakes

For a best estimate, Web CAD can be used for partial years when the date of diagnosis is more than 5 years after the start of intake. When the date of diagnosis is less than 5 years after the start of intake, IMBA (or an appropriate alternative) must be used to calculate the dose for a best estimate.

Over- and underestimates can be performed by extending the intake to a full first year or starting it in the following year, respectively.

4.3.2 Acute Intakes

Web CAD can be used as a best estimate for acute intakes when the date of diagnosis is more than 5 years after the intake date.

When the date of diagnosis is less than 5 years after the intake, over- or underestimates can be performed by entering an intake date of January 1 of the year of intake or the year after the intake, respectively.

5.0 DETERMINATION OF MISSED AND UNMONITORED INTAKE AND DOSE

Although referred to as dose in the following section, much of the discussion is focused on the *intake* calculation during the described periods. Doses from those intakes are assigned through the date of cancer diagnosis.

5.1 POTENTIAL FOR EXPOSURE

The presence of bioassay samples is often an indicator of potential for exposure, but if there are only baseline and termination samples (i.e., no other bioassay), they do not necessarily indicate a potential. Indicators of potential for internal radiation exposure include:

- Job title,
- Work location, and

- Changes in external dose. Note that measurable external dose alone is not necessarily an indicator of a potential for intakes. Material form is one factor (e.g., a sealed source might impart an external dose but, unless it is leaking, it is not an internal hazard).

The purpose of a bioassay measurement must also be considered in determining the potential for intake. A bioassay is occasionally performed for determining the background of a particular technique; such a measurement would not indicate a potential for intake. Prejob samples, taken before a particular work activity begins, are typically the equivalent of a baseline measurement. A site might also have a confirmatory monitoring program in which workers who are not expected to have an intake are sampled as confirmation that no intakes are occurring. Terminology and specific purpose vary, so this must be addressed on a site-specific basis.

Unless it can be shown that the site monitored all those with the potential for an exposure, lack of sampling for extended periods is an insufficient reason for assuming a change in exposure potential. If the three listed items do not change during a worker's employment history but there is information that indicates a potential for intake at some point (e.g., bioassay data or job title), and no information in the site profile to indicate otherwise, a potential for intake must be assumed for the entire employment period. For some claims, only the assignment of environmental intakes is appropriate. ORAUT-OTIB-0014, *Assignment of Environmental Internal Doses for Employees Not Exposed to Airborne Radionuclides in the Workplace* [ORAUT 2004a], provides guidance for determining such instances.

5.2 MISSED DOSE DETERMINATION

When bioassay results are negative, a constant, chronic intake is assumed throughout the possible exposure periods unless there are changes in the MDA as noted below or the bioassay was initiated in response to an identified incident (see Section 5.2.3). The date of the last sample result that is *less than the MDA* in the relevant period is used for the calculation. Do not use the date of a positive result and assume that no activity was detected. Missed and fitted assessments cannot be combined in a single IMBA run because different distribution assumptions are made. Note that, if a fitted dose has been calculated for all periods of potential intake, there is no missed dose.

Unless there is specific information that a person worked with known material types that differed throughout employment, fit only a single type at a time (i.e., do not mix types). Run each type separately and compare the total doses to determine the one that is most favorable to the claimant. Note that additional comparisons might be needed when there are multiple cancers. Refer to Section 7.3.2 for details.

As discussed in Section 3.4.6, chest counts cannot be used to assess Type F material. If there are only chest count measurements and Type F is applicable to the nuclide and site, Type F is considered unmonitored and Section 5.3 applies.

If the MDA changes during the potential intake period, the following must be considered in determining the chronic intake *for a given absorption type of a radionuclide* ("negative" here refers to a result ≤MDA):

- If the MDA *decreases* over time *and* the radionuclide reaches equilibrium slowly in the compartment of interest (e.g., Types M, S, or SS plutonium in urine, any Type S or SS material in the lung):

    1. Assume a single chronic intake for the entire potential exposure period.

2. Perform the fit using a result equal to MDA/2 on the date of the last negative sample. Only the lowest MDAs need to be considered in this scenario because any assessment of larger early values will result in the overestimation of the later, smaller MDA values.

- If the MDA *decreases* over time for radionuclides that reach equilibrium rapidly *or* if the MDA *increases* over time for any radionuclide, multiple chronic intakes should be used to determine the missed dose. Note that it might be necessary to deviate from this direction if later results are overpredicted. This is most likely to occur when there is a large drop in the MDA between consecutive samples or when samples are collected frequently throughout the employment period. To do this:

    1. Set the number of intakes to the number of periods of different MDAs in which the worker has bioassay results. Some variation of MDA within a period is acceptable; they do not have to be identical but should be similar. If the site profile lists a period with a specific MDA but the worker has no results during this time, this MDA is not considered in the analysis; the intake periods are based on the worker's sample dates.

    2. Start the first chronic intake period on the day exposure began and continue to the date of the last sample with the given MDA.

    3. Each following chronic intake is assessed from the day after the previous period to the date of the last sample in the next MDA period or to the last day of exposure for the final exposure period.

    4. Perform the fit using Measurement Value of MDA/2 on the date of the last sample in each period.

    5. Use the Uniform Relative error option with $K = 1$ for the error (this makes the error equal to the Measurement Value, which is MDA/2).

- When the MDA oscillates for an extended period, usually due to measurements with sample-specific MDAs, selection of the sample to use for the missed dose calculation can be claim dependent. For an overestimate the use of the largest MDA is appropriate and, conversely, the smallest value can be used for an underestimate. A best estimate depends on the pattern of the results, but in general a line that runs through the center of the MDA/2 values is suitable. See Figure 5-1 for an example. Contact the PIDS for assistance if necessary.

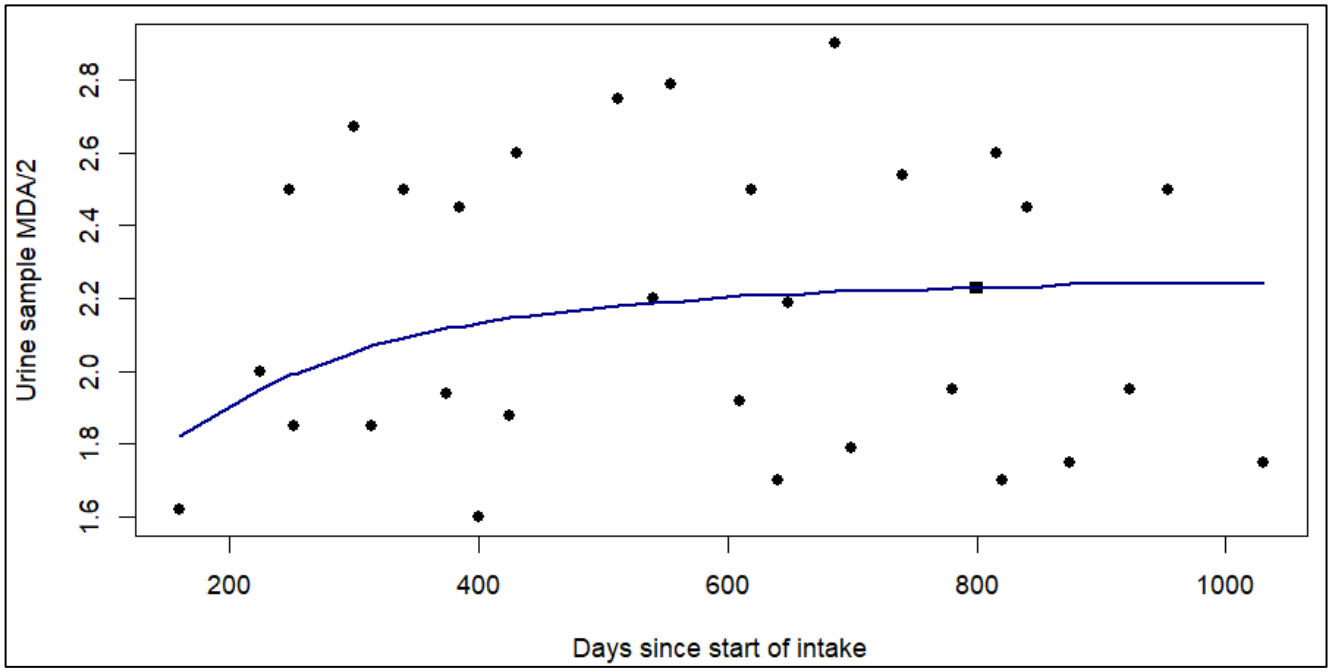


Figure 5-1. Example of a best fit for missed dose with an oscillating MDA. The result at 800 days, designated by the square, was used for the fit [ORAUT 2025a]. Attachment E contains an extended description.

Missed dose is entered into IREP using a triangular distribution and:

1. Lower bound (Parameter 1) = 0.
2. Mode (Parameter 2) = annual organ doses as calculated above.
3. Upper bound (Parameter 3) = 2 times the mode.

5.2.1 Multiple Bioassay Types

When there are multiple bioassay types available for a given nuclide, the missed dose is calculated using one bioassay type followed by a projection to the other type(s) to ensure that the results are not in disagreement. This is done by taking the intake calculated from the first bioassay type and using the “Intakes to Bioassay” tab in IMBA to calculate the expected excretion or retention on the dates of the second bioassay type. The projection is compared to half of the MDA of the second bioassay type. The intake calculated from the first bioassay type does not disagree with the second type if the projected values are less than or equal to half the MDA.

This section does not apply to fecal samples; see the next section for guidance on their use.

5.2.2 Fecal Samples

As discussed in Section 2.4.1.2, fecal bioassay is a sensitive method for assessing known or suspected acute intakes, and it can also be used for identifying long-term, low-level, or insoluble intakes. However, it is less useful for assessing missed dose from long-term chronic intakes because of the rapid drop-off of expected excretion rates as soon as the intake ceases. Although modeled as such for the dose reconstruction process, true chronic intakes are not common at most sites but are used to approximate the more likely scenario of a series of intermittent acute or short-term chronic

intakes, so the intake rate could be underestimated if the intake stopped a couple of days before sample collection.

Except for the case where a routine fecal bioassay program was in place and most of the results are positive (i.e., the worker likely had a chronic intake), fecal samples are not to be used for the assessment of chronic intakes, including missed dose. This uncertainty does not apply to acute intakes, so fecal results are to be used in those assessments. See Section 6.3.2 for further discussion on the assessment of a chronic intake based on positive fecal sample results.

Fecal bioassay is treated as follows for missed dose assessments:

- Use urine bioassay (and/or chest counts when available) to determine missed dose. Do not use the fecal results, and do not run projections to them (i.e., do not use them to limit the intake based on urine/chest counts).

- If the worker has only fecal results, assign a co-exposure intake in place of missed dose:

    - If all results are negative, nothing more needs to be done.

    - If some of the fecal results are positive, assess these as acute intakes and compare the doses on an annual basis to the co-exposure doses.

    - If there are no co-exposure intakes for the site in question, contact the PIDS for assistance.

- When associated with a documented incident (i.e., an acute intake on a known date), negative fecal results can be used to assess the missed dose associated with the known event. See Section 5.2.3.

- If there are scattered positive fecal results and negative urine results not associated with an incident:

    – Use the urine results to assess the missed dose. Do not project to the fecal sample results.

    – Use the fecal result(s) for a fitted dose assessment assuming an acute intake. Note that these do not need to be the same material type as the missed dose. In this case, project the fitted intake based on the fecal results to the urine sample results to ensure that they are not overpredicted. If they are, adjust the material type or intake date if unknown.

    – According to standard procedure, compare the missed and fitted doses on an annual basis and assign the larger of the two.

5.2.3 Missed Dose from an Incident

When a bioassay with a negative result was collected in response to an incident, a missed dose should be calculated assuming an acute intake on the date of the incident and half the MDA for the result. When assessing a missed acute intake for a best estimate, later results must not be in disagreement with the predicted values (i.e., results of later samples should not be overpredicted). Compare the resulting dose on a year-by-year basis to the chronic missed dose (if calculated). The assigned annual dose for a given year should be the maximum of the two values.

5.3 UNMONITORED DOSE

Unmonitored dose is the potential dose that could have been received by a worker but for which no monitoring of the worker was performed or monitoring data are not available. For unmonitored periods where there was a potential for intakes, the following priorities are used for assigning dose:

1. *Known ratio with other monitored nuclides.* For example, the contaminants in RU are not typically monitored directly but can be assessed based on a ratio to the calculated uranium intake.

2. *Co-exposure data.* Intake rates based on a compilation of data from monitored workers are available for specific sites, nuclides, and periods. Co-exposure intakes are discussed in more detail in Section 5.5. Note that in this step the only intake rates to be assigned are those covered by the intake periods included in the co-exposure study.

3. *Site-specific information.* Typically takes the form of default intake values in the site profile.

4. *Missed dose.* If monitoring is present before the unmonitored period, the missed dose intake rate can be extended beyond the last bioassay result. Note that this is still based on the actual last bioassay date and result; it does not mean to assume there was a later result. By extending this rate, an assumption is made that the intake rate was constant throughout the employment period.

5. *Extension of co-exposure doses.* If the above options are not available, it might be possible to extend the co-exposure intake rates beyond the modeled dates. Considerations include whether a best estimate is needed and the likelihood that the potential for intakes might have significantly decreased (e.g., from shutdown of activities or generally improved conditions in more modern times) or increased, as might occur with decommissioning activities. Contact the PIDS if there is no guidance in the site profile or dose reconstructor guidance documents.

5.4 MISSED VERSUS UNMONITORED DOSE

The line between missed and unmonitored dose is not well defined because material from an intake is excreted over an extended period depending on the half-life and retention characteristics of the nuclide. A long-lived, long-retained nuclide (e.g., plutonium or uranium) can be retained for decades with continuous excretion of small amounts. One result after many years of employment can contain activity from all previous intakes and provide information for determining an intake amount for all previous years and, in such a situation, a lack of bioassay samples for several years would not be considered unmonitored because an upper bound can be placed on the intake. This is not true for nuclides that are eliminated relatively rapidly from the body (e.g., ^{137}Cs and ^{3}H). An unmonitored period can precede a monitored period for these shorter-retained nuclides.

For both types of nuclides, a worker can be monitored for some period, after which there is an unmonitored period. The period after the last bioassay sample is considered unmonitored for both long- and short-retained materials.

A worker's bioassay data always take precedence over other data (e.g., co-exposure or site default values), unless the bioassay has been shown to be flawed or not representative of the worker's exposure.

For long-lived, long-retained radionuclides:

- Missed dose is calculated from the start of the potential intake period through the date of the last bioassay sample in the period being assessed. This period is considered to be monitored regardless of the date of the first bioassay sample.

- Unmonitored dose is assigned from the day after the last bioassay sample through the end of the potential exposure period.

- Long-lived, long-retained nuclides include all absorption types of plutonium, uranium, and americium, unless the only monitoring method is chest counting. Type M is not retained for significant periods in the lungs and the rules for short-retained radionuclides must be followed. As noted in Section 3.4.6, chest counting is not a valid method for monitoring intakes of Type F material.

For short-lived or short-retained radionuclides (including ^{137}Cs, ^{106}Ru, ^{144}Ce, and ^{90}Sr) during potential exposure periods:

- Missed dose is calculated in the intervals where there are bioassay results; other periods are considered to be unmonitored. Gaps of greater than 2 years between results are considered to be unmonitored.

- Unmonitored dose is assigned for the period up until 1 year before the first bioassay sample for the nuclide of interest.

- Missed dose is calculated from 1 year before the first bioassay result through the date of the last bioassay sample.

- If there are more than 2 years between two consecutive negative samples:

  – Missed dose is calculated through the date of the first of these samples.

  – Unmonitored dose is assigned from the day after the first sample until 1 year before the second sample.

  – Missed dose is calculated starting at 1 year before the date of the second result.

- Note that for very short-retained materials, notably tritiated water (HTO) and ^{131}I, these periods are shorter. Guidance on ^{3}H assessment is contained in ORAUT-OTIB-0011, *Technical Information Bulletin: Tritium Calculated and Missed Dose Estimates* [ORAUT 2004b]. In general, HTO exposure is assumed only during periods when bioassay samples were collected because it is inexpensive, easy, and quick. The primary exception to this rule is when a site began using HTO before the implementation of a bioassay program. For these two nuclides, missed dose can be calculated beginning 40 days before the first bioassay result. This is in agreement with the 40-day interval used in ORAUT-OTIB-0011.

- If unsure if a nuclide is considered short-retained, contact the PIDS.

5.5 CO-EXPOSURE DATA

Co-exposure dose is applied as a best estimate for workers with a potential for intakes of radioactive material who lack bioassay data or have unmonitored intervals. Data can be lacking because it was not available from the site or because monitoring was not performed. Typically, workers with a

significant potential for intake should be assigned doses at the 95th percentile with a constant distribution, while those with less potential are assigned the 50th percentile with a lognormal distribution. When co-exposure analyses and the site profile do not define how or to whom the intake should apply, “significant potential” is subjective, but in general it applies to people who were radiation workers with a potential for intakes of radioactive material. The dose reconstructor must make this decision based on the worker’s job titles and work locations, as well as other information in the claim file that could indicate a potential for intake. ORAUT-OTIB-0014 [ORAUT 2004a], provides guidance on job categories that are typically most likely to be in the upper end of the distribution. For sites that handled multiple independent sources of radionuclides, the site-specific TIBs or site profiles, where possible, provide guidance on which nuclides to assign. However, this can be a matter of dose reconstructor judgment, again, based on information in the claim file.

Co-exposure intake rates may be prorated when claim information clearly indicates intermittent exposure potential in any year of employment (e.g., a worker only entered an area with intake potential 1 day a week and was in a clean area all other days). In these situations, the co-exposure intake rate can be adjusted to only include the days with a potential for intake. For all other days, environmental exposures still must be considered. Note that co-exposure intake rates are based on calendar days, whereas occupancy can be based on workdays. This difference must be considered in any calculated adjustments. In addition, any adjustment for occupancy must be explained and documented in the dose reconstruction report. Contact the PIDS for assistance.

Note that the co-exposure intakes might end before a worker’s potential for exposure ends. This can be extended in some cases but other options for dose assignment take priority. See Section 5.3 for the list of the priorities.

5.6 EXAMPLE ASSIGNMENT OF CO-EXPOSURE AND UNMONITORED DOSES

Employment: 03/1/1957 to 07/12/1989

Job information: Production worker; single work location; no significant fluctuations in external dose results. The site profile provided no additional information on potential for exposure.

Bioassay: All results <MDA

MDAs remain constant over the employment period.

Plutonium-239

- Urine samples on 05/4/1960, 12/11/1963, 11/17/1980.
- Co-exposure intakes available through 12/31/1979.
- No site default intakes in the site profile.

Strontium-90

- Urine samples on 03/12/1965, 09/18/1965, 08/01/1966, 01/05/1978, 07/12/1978, 04/30/1979.
- Co-exposure intakes available through 12/31/1989.

Plutonium-239 dose calculation:

1. Calculate missed dose from 03/1/1957 through 11/17/1980 using 0.5 MDA on 11/17/1980.

2. For the period from 11/18/1980 through 07/12/1989, there are no site default values and no co-exposure intakes, so the missed dose is extended through this period.

Strontium-90 dose calculation:

1. Assign co-exposure data from 03/01/1957 to 03/12/1964.

2. Calculate missed dose from 03/13/1964 to 08/01/1966.

3. Assign co-exposure data from 08/02/1966 to 01/05/1977.

4. Calculate missed dose from 01/06/1977 to 04/30/1979.

5. Assign co-exposure data from 05/01/1979 to 07/12/1989.

6. If there are other nuclides associated with ^{90}Sr (as detailed in ORAUT [2015]), perform the above calculations first and assign the associated radionuclides from the resulting intakes in steps 1 through 5.

5.7 SHORT-DURATION MISSED DOSE

The fraction of an element that is absorbed into blood from the small intestine is defined by the parameter f1. A very small (i.e., $<1 \times 10^{-3}$) f1 value combined with a long half-life results in small fractions of material being excreted via urine. Because of the slow ingrowth to *urinary excretion* from chronic intakes, assessment of missed dose for nuclides with these small f1 values can result in implausibly large intake rates when the urine sample is collected shortly after the start of intake, particularly for those with relatively long half-lives. When calculating a missed dose where the only *urine sample* was collected within 1 year after the start of intake, the co-exposure intake rate should be assigned as a best estimate for nuclides with an f1 value <0.001 and a half-life greater than 50 years.

The most often encountered nuclides meeting these conditions are all material types of ^{238}Pu and ^{239}Pu. Note that no uranium material type has an f1 value less than 1×10^{-3}.

This is not to be confused with short-lived or short-retained materials, where “short” refers to the half-life or retention time in an organ rather than the intake period. It also does not apply to in vivo bioassay.

6.0 FITTING POSITIVE BIOASSAY RESULTS

This section describes the process for modeling intakes from positive results. Specific details of the mechanics of using IMBA are addressed in ORAUT-PROC-0002, *Use of Integrated Modules for Bioassay Analysis (IMBA)* [ORAUT 2003]. When both are appropriate, these “fitted doses” are combined with the separately calculated missed dose. Additional guidance on fitting data for a best estimate are in Section 7.3.

6.1 OVERVIEW

The fitting of bioassay data to an intake is a somewhat subjective process, particularly when dealing with historical data because intake dates are frequently unknown and additional follow-up sampling is not possible. Fits should be as simple as possible; no more complexity than necessary should be applied to a given claim. This means if a quick and simple over- or underestimate can be performed using the bioassay data (see Section 7.0 for discussion), no further fitting should be tried.

In general, the overall pattern of the data should be fit rather than each individual result. A “good fit” is one where a minimum number of negative results are overpredicted (i.e., the predicted result exceeds the MDA) while a minimum number of positive results are substantially under- or over-predicted.

It is *not* realistic to develop an intake scenario that yields predicted results that are identical to the measured values for all or even most of the measurements because the retention and elimination of radioactive materials, as well as the measurement of the material, are stochastic processes that result in statistical variations. For urine samples, the concentration varies even within 1 day. In addition, an exact match to each measured result is often achieved only through a set of very unrealistic assumptions that are often not favorable to claimants. An example of this is fitting each positive result to a separate intake. This often requires the assumption that each intake occurred only 1 or 2 days before the bioassay sample. If the samples were collected as part of the routine bioassay program (as opposed to incident-related samples), it is unlikely that the program caught each unsuspected intake immediately upon occurrence. Known intake dates should be used when available.

6.2 GENERAL GUIDELINES

Assessment of positive bioassay results is subjective in the absence of known intakes, so the following guidelines are provided for performing a best estimate:

- Use all positive bioassay results.
- For results that are not positive:
  - Enter the MDA value for both the Measurement Result and Measurement Error.
  - Set the Data Type of the first negative result after *each* positive or set of positive results to **<LOD**.
    - If there are several consecutive positive results, label no more than two of the immediately after consecutive negative results as **<LOD**.
    - For fewer than five consecutive positive results, include only the first immediately after a negative result. Use of additional <LOD results, particularly for chronic exposures, frequently yields a fit that appears to underestimate the general trend of the data.
  - All other negative results should have a Data Type of **Excluded**.
  - Note that the presence of a negative result does not mean that a new intake must be assigned for the next positive result.
- Although not necessarily used in fitting, all results (including those that are excluded) should be plotted on the graph because they must be consistent with the final fit.
- Fit all the results simultaneously (i.e., a single IMBA run), even if there are multiple intakes. A mix of chronic and acute intakes can be applied, as can a single or multiple chronic intakes. A single chronic intake can also be fit when there are only intermittent positive results that are relatively small (e.g., within a factor of 2 of the MDA); this could be representative of a low-level chronic intake just below the MDA. Note that the limitations on the use of <LOD apply here as well.

- For positive results, see Section 4.1.2 for guidance on assigning the measurement error. Error selection can have a significant impact on the resulting fits.

- If there are several negative results or stretches of employment where all results are negative, missed dose is assessed separately. See the discussion in Section 5.2.

- Use known information about intakes where available (e.g., intake date, material type, and particle size distribution). For unknown parameters, begin with default values where possible. These can be adjusted as necessary, but there must be sufficient justification when doing so.

  For an unknown intake date, the default is the midpoint between the date of the positive result and that of the previous sample measuring the same radionuclide from the same bioassay type. Intake dates should only be varied if projections from the intakes are inconsistent with later data of any bioassay type (e.g., several negative results are predicted to have had detectable levels of activity). As discussed in Section 6.1, it is neither necessary nor desirable to obtain an exact fit to each result because variation in excretion rate is to be expected.

- If the material type is unknown, perform a fit for each possible type. Select the one that yields the largest total *dose* to the applicable organ for the years of interest. Note that the largest intake does not necessarily correlate to the largest dose. In some instances, it might be possible to rule out a material type based on later, overpredicted sample results or disagreement with other measurement types. If one type provides an unarguably better fit, use it. This can generally be shown only where:

  – A single intake has many (more than 10) consecutive positive results,

  – There are contemporary data (later than 1989) associated with intakes 25 years or more earlier (depending on the nuclide and its associated half-life), or

  – Results from other bioassay methods cannot be reconciled with the larger dose determinations (e.g., the intake determined from urine samples predicts detectable activity in a chest count but all results are <MDA; in this case, the material type that yields this larger intake would be ruled out).

  – Note that in the case of multiple acute intakes, it is not necessary to assume the same material type for all intakes. A mix might be necessary to best fit the data.

- If most results are positive and scattered throughout the intake period (with no more than a few consecutive negative results), use all results for the intake assessment. For those few results that are negative, enter the MDA for the value and use Data Type **<LOD**. Note that the issue of measurement error (Section 4.1.2) also applies here.

6.3 FECAL SAMPLES

While there is additional uncertainty associated with fecal samples and they are therefore not to be used for assessing missed dose (see Section 5.2.2), a positive result is a possible indicator of an occupational intake and must be addressed.

6.3.1 Acute Intakes

Fecal samples are sometimes collected in response to an incident where an intake is suspected because they can be more sensitive than other monitoring methods when collected soon after intake. For known intake dates, both positive and negative fecal sample results provide useful information

and should be used in conjunction with other results in the intake assessment. Contact the PIDS for assistance with reconciling results of multiple monitoring methods.

6.3.2 Chronic Intakes

Some sites instituted a routine fecal sampling program because of its sensitivity, particularly for uranium and transuranic nuclides. This can result in multiple sequential positive fecal samples at sites where chronic uranium intakes were not uncommon. This is further complicated by the presence of environmental uranium because a larger fraction of an ingestion intake is excreted via the fecal pathway than that of inhaled material.

Contact the PIDS for assistance when this situation is encountered.

7.0 OVERALL ASSESSMENT METHODS

There are typically several approaches that can be applied to a given claim. The best approach is that which takes the least amount of time while still producing the correct decision. Many claims do not require a detailed, precise dose assessment; efficiency methods can be used to expedite claim completion with sufficient levels of precision to allow the U.S. Department of Labor (DOL) to arrive at correct compensation decisions. There are two general types of expediting methods that can be applied: overestimates and underestimates. When neither of these can be applied to a claim, a more refined best-estimate assessment is needed.

7.1 OVERESTIMATE

An overestimate is the assignment of an intake or dose that exceeds the possible exposure of the worker. If the resulting POC, including all sources of potential exposure, is less than 45% (note that this value is determined by Project Management and NIOSH and is subject to change), further refinement is not necessary because it would only lower the assigned dose.

This method is typically appropriate to cancers of nonmetabolic organs because the radioactive material does not concentrate in such organs. Therefore, relatively large intakes can yield small doses. The method also lends itself to the development of generic values that can be used for many workers. Individual overestimates can also be made using worker-specific information.

7.1.1 Generic Overestimates

Several methods have been developed and documented in TIBs.

Most of the overestimating methods are applicable to workers with no positive bioassay results. However, this can be extended to workers with positive results as long as the positive results are taken into account (i.e., it is shown that the assigned intake yields larger projected values than those reported, or the positive results are assessed separately and the subsequent dose is added to the efficiency method results).

7.1.2 Worker-Specific Overestimates

Overestimates can sometimes be applied to workers with positive bioassay data. In such cases, most of the bioassay results should be overpredicted by the selected intake. This can be done by running a chronic intake assessment using only the largest bioassay result; all others should be plotted but excluded from the fit. If there are several large results, use of the earliest value to perform the fit typically yields the largest intake. After calculating the intake, review the measured (Measurement

Rate) versus predicted (Theoretical Rate) results to determine if most results have been overpredicted (this can be done quickly with the graph).

If there are later results that are underpredicted, determine the ratio of the measured result to the predicted result, multiply the intake rate by this ratio, then run the Intakes-to-Bioassay calculation to demonstrate that all bioassay results have been overpredicted. Assign the adjusted intake rate (i.e., original intake rate times the ratio) that overestimates the later results.

A similar method can also be used when there is an acute intake. Start by using only one result and adjust the intake as necessary to obtain an overestimate of all the results associated with the intake.

The constant distribution can be assigned in IREP because this is an upper bound of the dose [NIOSH 2002].

7.2 UNDERESTIMATE

An underestimate is the assignment of a dose to a worker that is less than the dose that would potentially be assigned under this program. If the resulting POC is greater than 52% (note that this value is determined by Project Management and NIOSH and is subject to change), further refinement is not necessary because it would only increase the assigned dose. An underestimate is typically performed in the form of a partial assessment of dose, such as reconstruction of a single incident or missed dose only. The assigned distribution in IREP depends on the type of dose (e.g., missed or fitted) that is calculated.

An underestimate is most likely to be appropriate when applied to metabolic organs, particularly in claims where the detection level for the nuclide is large. This is frequently the case with actinides in the earlier decades of the DOE complex. In such instances, a missed dose calculation alone might be adequate for determining compensability.

Because this method is dependent on a worker's bioassay data, the details are claim specific and do not lend themselves to a generic approach that can be documented in a TIB.

7.3 BEST ESTIMATE

A best estimate is required when an efficiency method results in a decision that is incompatible with the assumptions (i.e., an underestimate yields a POC less than 52% and an overestimate yields a POC greater than 45%; as noted previously, the specific values are subject to change). The purpose of this Project is to provide dose reconstructions with sufficient levels of precision to allow the DOL to arrive at correct compensation decisions. A best estimate is based on all available data and is the most realistic assessment that can be performed with these data and the requirements of the Project.

When information for a particular parameter value is unknown or there are multiple options, the choice that is favorable to claimants (i.e., the one resulting in the largest POC) is selected.

7.3.1 Performing a Best Estimate Using Bioassay Data

A best estimate uses all available information. Both missed and fitted dose are included but are assessed separately and compared. Fitting them simultaneously results in the inappropriate combining of distributions (lognormal and triangular) in a single fit and possible underestimation of both the individual fitted and missed doses. The exception to this is when the vast majority of results

are positive, as discussed in the final bullet of Section 6.2. In such instances, there is no missed dose because all the results and intake periods are taken into account in the fitted dose.

1. Calculate the missed dose:

   Ignore positive data (this means that the date of the last result <MDA is used for the missed dose calculation) and perform a missed dose (mode only) calculation as described in Section 5.2. If multiple material types are possible, select the one that yields the largest total dose to the applicable organ for the years of interest. If there are no (or very few, as noted above) results <MDA, no missed dose is calculated.

2. Calculate the fitted dose:

   Fit the positive data in accordance with Section 6. Unless known intake dates are documented or bioassay results are indicated to be special rather than routine, it is not considered a best estimate to assign all intake dates at 1 to 2 days before the date of a positive bioassay sample. As noted above, this is not a realistic scenario and in most cases is not favorable to the claimant. If this is the only way a fit can be obtained, it is likely that an inappropriate material type is being applied or too much effort is being made to fit every result exactly.

3. Assigned annual dose for a given year is the maximum value from step 1 or 2. Choose the IREP annual dose distribution type based on missed or fitted dose assignment:

   a. For years in which the dose determined in step 1 is larger than that from step 2, use the triangular distribution, where Min = 0, Mode = annual dose, and Max = Mode*2.

   b. For years in which the dose from step 2 is equal to or larger than that from step 1, use the lognormal distribution, where Median = annual dose and the GSD = 3.

7.3.2 Multiple Cancers

Consistent assumptions must be made for all cancers when performing a best estimate, e.g., material types, radionuclides. Selections are based on those parameters that yield the largest overall POC.

Examples include:

- The material type for a given intake (e.g., missed, fitted) of a radionuclide must be the same for all organs. A consistent material type must be used across all organs for each fitted dose for a given nuclide and similarly, a consistent material type must be used for the missed dose calculation for all organs.

- When using the Chooser tool for selecting the radionuclide yielding the largest dose from a whole body count result, the same nuclide must be selected for all cancers when a best estimate is required.

- If a bioassay sample was analyzed using a nonspecific technique (e.g., gross alpha, gross beta), the same nuclide must be assumed for all cancers.

An exception to this rule is the assessment of doses using ORAUT-OTIB-0054, *Fission and Activation Product Assignment for Internal Dose-Related Gross Beta and Gross Gamma Analyses* [ORAUT 2015]. A number of scenarios comparing the largest and smallest doses to various organs from the ORAUT-OTIB-0054 mixture were reviewed. Because the differences were not large and because of the massive amounts of computing time required to make material types consistent across organs, it

was determined to be acceptable to run each organ separately for ORAUT-OTIB-0054 claims with multiple cancers, using the maximum material types for each organ.

7.4 WORKING WITH MULTIPLE BIOASSAY TYPES

7.4.1 Fitting Multiple Bioassay Types

While IMBA allows the simultaneous fitting of multiple bioassay types, this option must be applied with extreme caution. Oftentimes different bioassay techniques yield results that are of different orders of magnitude, so one of them can completely dominate the fit regardless of how sensitive or accurate the relative techniques are. The preferred approach for most claims, particularly those where most or all results from one technique are ≤MDA, is to fit one of the bioassay types and project the resulting intake to the dates of the second technique to make sure the two are not in disagreement. The most appropriate scenario for assessing them simultaneously is the case of a known acute intake with primarily positive results for all assessment techniques.

Multiple bioassay types are commonly seen for sites with plutonium mixtures, where plutonium is measured directly in the urine and ^{241}Am is measured in the lungs to determine the plutonium intake. In this case, because two different nuclides are measured, they cannot be combined into a single assessment. Section 9.1.1 provides additional information on plutonium assessments using urine samples and chest counts. Multiple bioassay methods are also often seen at uranium facilities where, in addition to routine urine samples, chest counts are sometimes used to determine if there is long-term buildup in the lungs.

7.4.2 Projections

As described in the previous section, when there are two bioassay types available for a given nuclide, the assessment is typically calculated using one bioassay type followed by a projection to the other type to ensure that the results are not in disagreement.

When missed dose is being calculated, the projection should be compared to half of the MDA of the second bioassay type. That is, the intake calculated from the first bioassay type does not disagree with the second type if the projected values are less than or equal to half of the MDA.

For fitted doses, projections are compared to the full MDA because the fits are based on values exceeding the MDA.

8.0 RADIATION TYPE

IREP requires a Radiation Type to be associated with each entered equivalent dose. The radiation effectiveness factor (REF) is analogous to relative biological effectiveness in radiobiology. In IREP, the equivalent dose to an organ is converted to absorbed dose and modified with the appropriate REF as part of the POC calculation. The selected REF depends on the type of radiation, of which there are six that are applicable to internal dose:

- Alpha,
- <15-keV electrons,
- >15-keV electrons,
- <30-keV photons,

- 30- to 250-keV photons, and
- >250-keV photons.

Note that >250-keV photons and >15-keV electrons are the reference radiations and have a point value REF of unity (i.e., the categories are interchangeable). The REFs for the other radiation types are expressed as distributions [Kocher et al. 2002].

Many radionuclides emit multiple types and energies of radiation that encompass more than one REF category. Each radionuclide was assigned to a single REF category associated with the majority of energy emitted by the radionuclide. Radiation weighting factors were also considered in categorizing the radionuclides. Attachment D contains the values to be used for all nuclides available in IMBA and the Project-developed tools.

Radionuclides that do not emit alpha radiation but have progeny that do are assigned to the alpha category because it is favorable to claimants. The single exception is ^{147}Pm, which is assigned to the "electrons E >15-keV" radiation category because of the extremely long half-life of the ^{147}Sm progeny.

9.0 NUCLIDE SPECIFIC ISSUES

9.1 PLUTONIUM

9.1.1 Plutonium-239 Mixtures

Plutonium-239 is found in various mixtures depending on the purpose of the material. This typically includes several plutonium isotopes as well as ^{241}Am from ^{241}Pu decay. A given bioassay technique does not necessarily measure all the components; different methods can be used to measure the different nuclides. There are two primary complications in assessing intakes of these mixtures: (1) ^{241}Am activity increases over time while the plutonium activities are decreasing, which means that the ratios are not constant and makes the age of the material a factor; and (2) an assumption that is favorable to the claimant for one technique might not be for another technique. Dose reconstructors must therefore be sure to take all information into account.

Note that if the material is from a plutonium heat source, the primary plutonium isotope is ^{238}Pu and this discussion is not applicable.

9.1.1.1 Background Information

Plutonium mixtures are characterized by their ^{240}Pu content; they are referred to by its weight percentage. Weapons-grade mixtures are 6% by weight ^{240}Pu while fuel-grade plutonium is 12% by weight ^{240}Pu.

Americium-241 builds up from near zero at the end of irradiation; however, it is removed during separation of the plutonium product and begins to build up again as the ^{241}Pu remaining in the product decays. Therefore, the ratio of ^{241}Pu to $^{239+240}$Pu decreases from the time of the end of irradiation because of decay (^{241}Pu has a half-life of only 14.4 years); whereas the ratio of ^{241}Am to $^{239+240}$Pu increases from the time of the last separation of the ^{241}Am from the plutonium.

Because ^{241}Am emits a gamma ray that can be detected more readily than the emissions from any of the plutonium isotopes, chest counting for its presence is sometimes performed and used as an indicator of plutonium intake. Interpretation of the result is complicated by the simultaneous production (from the decay of ^{241}Pu) and decay of ^{241}Am. See Section 9.1.1.3 for guidance on using IMBA for this assessment.

9.1.1.2 Assumptions for Use in Dose Reconstruction

As with all dose reconstructions, when there is known information about an intake it should be used regardless of default assumptions. In this instance, if the plutonium mixture or age of the material is known, it should be used rather than the information below.

9.1.1.2.1 Mixture of Material

If the mixture is unknown and the intake is being calculated from urine sample results, the 12% mixture may be used as a default assumption that is favorable to claimants.

Because there is less ^{241}Am in the 6% mixture than the 12% mixture, use of a 6% mixture is the default starting point for limiting doses based on chest counts.

If both types of data are available, it is necessary to compare them. For likely noncompensable claims, it is acceptable to overpredict one of the sets of monitoring results. However, if the claim is likely compensable, the selected intake scenario must not contradict any of the worker's monitoring data (urine or chest counting). For example, if the intake is based on urine samples with fuel-grade plutonium (12% mixture), run a prediction to the dates of the chest counts based on the intake (using the Intakes to Bioassay tab in the IMBA Bioassay Calculations window). If the predicted values are greater than the measured values of ^{241}Am in the lung including ingrowth from ^{241}Pu, the intake can be used for a likely noncompensable claim but not for a likely compensable claim. In the latter case, the intake would then be determined using the 6% mixture assumptions to fit the chest count results.

9.1.1.2.2 Age of Material

As noted in Section 9.1.1, ^{241}Am activity increases over time while the plutonium activities are decreasing. Because of this, the time since the end of irradiation or last separation, referred to as the age of the mixture, is needed to accurately assess the ^{241}Am content, particularly when chest counts are present. Default age assumptions are given below. Site-specific information takes precedence over these assumptions.

If information about the material age is not available, the following assumptions are applied. These assumptions can be used for either urinalysis or chest counting.

- Until the fifth year of site operation, assume fresh plutonium.
- For years 5 through 9, assume a 5-year-old plutonium mixture.
- After these times, assume 10-year-old plutonium.

9.1.1.3 Americium Ingrowth to Plutonium Mixture

As noted above, interpretation of chest count results for ^{241}Am in a plutonium mixture is complicated because it is continually produced by the decay of ^{241}Pu. IMBA has a module that allows this to be modeled.

Initial set-up steps:

1. Open new session of IMBA.

2. In the Indicator Radionuclide box:

    a. Select **Americium-241**.

    b. Change the Number of Associated Radionuclides to **1**.

3. In the Associated Radionuclides box:

    a. Select **Plutonium-241**.

    b. Set the Abundance for the associated radionuclide. This value is 100 times the ratio of activity of ^{241}Pu to activity of ^{241}Am in the mixture. For example, if the ratio of ^{241}Pu to ^{241}Am activity is 49.1, the Abundance is 4910.

4. From the toolbar at the top left of the screen, select **Advanced**, **Advanced Dosimetry Options**.

    a. Select **Bioassay** tab.

    b. Check **Allow ingrowth of Am-241 from Pu-241.**

    c. Select **Pu/Am ratio fixed at start of each intake regime** (this should already be selected as the default setting).

    d. Click **OK**.

5. In the Units block:

    a. Select appropriate units for Intake.

    b. Set the Reference Date in the Specify Time As section.

    NOTE: This date *must* be identical to the specified date for the beginning of the intake regime.

    c. Click the **Date** radio button.

6. Select **Acute** or **Chronic** as appropriate in the Intake Regimes block.

7. Set the Start Date for IR1 (must be identical to the Reference Date entered in step 5b).

8. Set the End Date for IR1 (if the intake is chronic).

9. Load the ICRP Default f1 value and Absorption Type. For a Type S plutonium mixture, select the **ICRP71 Type S** option for ^{241}Am.

To determine the ^{241}Am content in the lungs at any time after intake with a known initial intake of ^{241}Am and ^{241}Pu from plutonium urine results:

1. Enter the ^{241}Am intake activity on the Main page (IMBA automatically calculates and includes the associated ^{241}Pu for this intake per the Associated Radionuclide abundance set above).

2. Click **Bioassay Calculations**.

3. Set up a table for lung data and enter the date(s) for the required projected lung counts (green area for projections).

4. Use the Intakes to Bioassay tab to calculate the lung count value on the specified dates.

5. Use the projected values to compare to actual lung count data.

To calculate the ^{241}Am and ^{241}Pu intake from lung count data:

1. Click **Bioassay Calculations** on the Main screen.

2. Set up a table with the appropriate lung count data (blue area for data).

3. On the Bioassay to Intake tab, check **Lungs**.

4. Click **Start Calculation**.

5. Go back to the Main screen to see the calculated ^{241}Am intake.

6. Use the ^{241}Am intake value and the activity ratio of ^{241}Pu to ^{241}Am to determine the ^{241}Pu intake.

7. Use the activity ratios for ^{238}Pu and ^{239}Pu to ^{241}Am to calculate the ^{238}Pu and ^{239}Pu intake values for the mixture.

9.1.2 In Vitro Bioassay Techniques

9.1.2.1 Gross Alpha Measurements

When a gross alpha measurement is used to analyze plutonium in an in vitro sample, both ^{238}Pu and ^{239}Pu activity will be included in the result, as well as ^{240}Pu. Plutonium-241 is a beta emitter, so its activity will not be included. Other nuclides might be present depending on the chemistry used in preparing the sample.

9.1.2.2 Alpha Spectroscopy

Plutonium-239 and ^{240}Pu emit alpha particles of almost identical energies, so they are indistinguishable when analyzed by alpha spectroscopy. A result reported as ^{239}Pu includes both isotopes unless the site has done a calculation to remove the ^{240}Pu activity (this is uncommon).

9.1.2.3 Thermal Ionization Mass Spectrometry

Thermal ionization mass spectrometry, commonly referred to as TIMS, is more sensitive than alpha spectroscopy for ^{239}Pu, so it is used at some sites for its analysis. Other isotopes, such as ^{240}Pu, would not be included as part of the measurement and would be reported separately.

9.2 URANIUM

9.2.1 Mixtures

9.2.1.1 Natural and Enriched Uranium

Uranium intakes are always comprised of a mixture of isotopes and often contain other radionuclides. Natural uranium, sometimes referred to as normal uranium, consists of ^{234}U, ^{235}U, and ^{238}U. Mixtures

might also be depleted (DU) or enriched (EU), where “depleted” and “enriched” refer to the ^{235}U mass content of the mixture relative to natural uranium. The degree of enrichment or depletion varies with the specific process used for separations. Conventional enrichment techniques result in increased amounts of both ^{234}U and ^{235}U, while that enriched by atomic vapor laser isotopic separation (AVLIS) technology has far smaller quantities of ^{234}U. AVLIS is not an extensively used technology, and its use is noted in the site profile. Example ratios are shown in Tables 9-1 and 9-2. Note that there is a significant difference in ratios depending on whether content is measured as mass or activity. This is due to the much larger specific activity of ^{234}U relative to the other isotopes. Conversion of reported results to a quantity useful for intake and dose assessment depends on the measurement method and enrichment.

Table 9-1. Example percent uranium content by mass (%).[a,b,c]

Nuclide	Natural[d]	DU[e]	LEU[e]	HEU[e]	AVLIS 5% enriched[f]
U-234	0.005	0.001	0.029	1.06	0.00006
U-235	0.72	0.20	3.5	93.5	5.00
U-236	0	0.0003	0	0.21	0
U-238	99.3	99.8	96.5	5.27	95.0

a. HEU – highly enriched uranium; LEU – low-enriched uranium.
b. Enrichment fractions are examples and vary by site depending on process.
c. Fractions do not necessarily sum to 100% due to rounding.
d. ORAUT [2025c].
e. Example enrichment mixtures from IMBA.
f. Mansfield and Wood-Zika [2000].

Table 9-2. Example percent uranium content by activity (%).[a,b,c]

Nuclide	Natural[d]	DU[e]	LEU[e]	HEU[e]	AVLIS 5% enriched[f]
U-234	49	15.5	81.8	96.8	45.0
U-235	2	1.07	3.44	2.97	13.9
U-236	0	0.05	0	0.20	0
U-238	49	83.4	14.7	0.026	41.1

a. HEU – highly enriched uranium; LEU – low-enriched uranium.
b. Enrichment fractions are examples and vary by site depending on process.
c. Fractions do not necessarily sum to 100% due to rounding.
d. ORAUT [2025c].
e. Example enrichment mixtures from IMBA.
f. Mansfield and Wood-Zika [2000].

9.2.1.2 Recycled Uranium

Uranium might also be recycled. RU in the DOE complex is uranium that has been irradiated in a reactor, recovered by one of four primary chemical processing sites, and returned to facilities that further processed the uranium for feed materials preparation, purification, enrichment, uranium parts fabrication, etc. This RU from reactor plutonium production targets and fuel elements contains trace quantities of transuranic elements (primarily ^{238}Pu, ^{239}Pu, ^{237}Np, and ^{241}Am) and fission products. Technetium-99 was the predominant fission product, although other isotopes were also present (short-lived such as $^{95}Zr/Nb$ and $^{106}Ru/Rh$ for material processed immediately and longer-lived such as $^{90}Sr/Y$ and ^{137}Cs) when the materials left the primary chemical processing plants. Plutonium production reactors produced most of the RU, although the uranium in spent fuel from test reactors and military power reactors was also recycled. The RU period began in 1952; Hanford was the first production site to recover and ship uranium from spent fuel [DOE 2003].

Site profiles provide information for assigning these contaminant radionuclides, typically based on a ratio to the uranium content. The ratio might be based on uranium mass, total uranium activity, or a

single uranium isotope. Section 3.3.3.1 provides guidance on the selection of absorption type for RU contaminants.

9.2.1.3 Uranium-233

Uranium-233 might be present at some sites as a fraction of a mixture of other nuclides and uranium isotopes or as the primary component of a source. It is indistinguishable from ^{234}U by alpha spectroscopy so there might be some bioassay reports listing a result as both isotopes, e.g., $^{233/234}$U. The ^{233}U committed dose conversion factors are somewhat larger than those for ^{234}U, ranging from about 3% for most organs from a Type F intake to a maximum of 55% to the bone surface from Type S.

A ^{233}U source often has a ^{232}U contaminant component. This is typically very small but given that the committed organ doses for ^{232}U are 1.6 to 47 times larger than those for ^{233}U and the specific activity of ^{232}U is 3 orders of magnitude larger than that of ^{233}U, a few percent of ^{232}U *by mass* in a ^{233}U mixture can yield a noticeable increase in dose.

9.2.2 Methods for Assessing Isotopic Uranium Results

Note: The following guidance does not apply to $^{233/232}$U mixtures because of the items discussed in Section 9.2.1.3.

Uranium derived from natural uranium (i.e., natural, depleted, and enriched) is comprised of several isotopes, all of which must be accounted for in a dose assessment. When a urine sample is analyzed for uranium using a gross alpha technique, the measured activity is representative of the total uranium activity because all isotopes are alpha emitters of relatively similar energies. When uranium is analyzed isotopically via alpha spectroscopy or inductively coupled plasma-mass spectrometry (ICP-MS), isotopes are reported separately and the individual components must all be accounted for in the dose assessment because this is a known mixture of material. That is, assessing only one or two of the components, even if they are >MDA and the others are not, results in an underestimate because at a minimum ^{234}U, ^{235}U, and ^{238}U are present in a mixture; ^{236}U is also present in some RU mixtures. One or more isotopes might also be included in an in vivo bioassay report.

There are several options for assessing an isotopic uranium result. The selected method is dependent on claim details.

9.2.2.1 Alpha Spectroscopy

The general rule of thumb for all claims is to start with the isotope with the highest activity fraction in the mixture.

9.2.2.1.1 Underestimate

Assessment of a single isotope can be used as an underestimate for a likely compensable claim. The appropriate IREP distribution is dependent on the type of intake that was calculated:

- Triangular for missed dose, and
- Lognormal for fitted dose.

9.2.2.1.2 Overestimate

For an overestimate, the activity from all three (four, if ^{236}U is included) isotopes can be summed and assessed as ^{234}U (^{238}U may be used for DU). For isotopes <MDA, add the MDA value (all MDAs can be summed for an overestimated missed dose).

Run the intake and dose estimates using the MDA values rather than 0.5 MDA.

Use a constant distribution for the doses in IREP.

9.2.2.1.3 Best Estimate

For a best estimate, all isotopes of uranium must be included in the appropriate proportion for the site's enrichment. Default enrichments are documented in the site profile. Before an assessment can be done, the isotopic results must be adjusted to a total uranium activity value:

1. If all isotopes for a given sample are >MDA

    a. Sum the individual results.

    b. This is a claimant favorable approach because there is unlikely enough intake mixture or background information to adjust the measured results. No further adjustments are needed.

2. If ^{234}U and ^{238}U >MDA:

    a. Sum the ^{234}U and ^{238}U results.

    b. Adjust the result to obtain total intake activity by summing the fractions (based on the appropriate site values) that ^{234}U and ^{238}U contribute to the total uranium activity and dividing the result by this value.

    For example, natural uranium consists of 49% ^{234}U, 49% ^{238}U, and 2% ^{235}U. Given 0.5 dpm ^{234}U and 0.5 dpm ^{238}U in a sample with ^{235}U <MDA, the total activity would be:

    $$(0.5 + 0.5) \text{ dpm}/(0.49 + 0.49) = 1.02 \text{ dpm} \quad (9\text{-}1)$$

3. If only a single isotope is >MDA (for alpha spectroscopy results this could be ^{234}U or ^{238}U), use the positive result to determine the total uranium activity. The calculation is the same as the example in method 2 above but based on the single positive isotope values. For example, given an EU mixture consisting of 82% ^{234}U, 15% ^{238}U, and 3% ^{235}U with a positive ^{234}U result of 0.5 dpm, the total activity is:

    $$0.5 \text{ dpm}/0.82 = 0.61 \text{ dpm} \quad (9\text{-}2)$$

4. If no isotopes >MDA, this calculation is the same as that in method 3 but is based on a single MDA. For DU, use the ^{238}U MDA and fraction; for all other enrichments use ^{234}U values. Note that this applies to all results with no positive isotopes regardless of whether a missed or fitted dose is being assessed.

Once all results have been adjusted, they are assessed as usual. The total activity is assessed in IMBA and Web CAD as ^{234}U, with the exception of DU, which is assessed as ^{238}U. Results that had

any component exceeding the MDA are treated as positive values and those with all components less than the MDA are treated as <MDA.

Note that for a best estimate it is not appropriate to sum the components if any are <MDA because the isotopes will not be in the correct ratios. This is demonstrated with an example:

- When using alpha spectroscopy, the MDAs for ^{234}U, ^{235}U, and ^{238}U are often similar. For this example, assume that each MDA is 0.5 dpm.

- Given the EU mixture in item 3 above, the ^{234}U activity is 5.5 times that of ^{238}U and 27 times ^{235}U. Given a ^{234}U MDA of 0.5 dpm, the total uranium activity would not be expected to exceed 0.5 dpm/0.82, or 0.61 dpm, where 0.82 is the ^{234}U fraction of the total.

- The ^{238}U activity fraction of the total activity is 0.15, so the ^{238}U activity would not be greater than 0.61 dpm × 0.15, or 0.09 dpm.

- Assuming all components could be as large as 0.5 dpm (i.e., applying the same MDA to each of them) and summing them would overestimate the total activity.

- *This is why the isotope with the largest fraction in the mixture is used for determining the total activity*.

9.2.2.2 Inductively Coupled Plasma Mass Spectrometry

This analytical technique measures the mass of a nuclide rather than activity. This is a relatively inexpensive and quick method for analyzing uranium and has been implemented at some uranium facilities starting around 2000. Oftentimes only ^{235}U and ^{238}U are reported because ^{234}U has a large MDA and contributes a very small mass to all uranium mixtures, which means it is not likely to be detectable. This does not mean that it does not impart any dose. As shown in Table 9-2, ^{234}U contributes a significant fraction of the total activity, so it must be included in the assessment. This also means that a uranium enrichment assumption is necessary for assessing most claims analyzed with this method.

The site profile should include a description of the measurement techniques used at the site. If no methods are specified and reported uranium results include only ^{235}U and ^{238}U, or the ^{234}U MDA is much larger than those of the other isotopes, then it is likely that ICP-MS was performed. Regardless of the technique, ^{234}U should be included in the intake and dose assessment.

While the methods for determining the total activity are similar to those used for alpha spec, the overestimation method described in Section 9.2.2.1.2 cannot be used if there is no reported ^{234}U value. To assess the dose in this case, conversion to total activity based on uranium enrichment will be needed. Once the appropriate enrichment has been determined, convert the result to total uranium activity.

Uranium-238 accounts for most of the mass except for the case of highly enriched uranium (HEU), so it is typically used to perform the conversion. Use the ^{238}U result (or MDA, if the result is negative) to obtain the total uranium activity. This is used even if there are negative ^{234}U results because of its very small contribution to total mass and relatively large MDA, so use of its MDA would yield a disproportionate contribution from ^{234}U.

Using natural uranium as an example once again, the composition by mass is 0.0054% ^{234}U, 99.27% ^{238}U, and 0.72% ^{235}U. Given a ^{238}U result of 6 µg, the total activity in the sample is:

$$\frac{6\ \mu g}{0.9927} \times 6.85 \times 10^{-7}\ \mu Ci/\mu g = 4.14 \times 10^{-6}\ \mu Ci \quad (9\text{-}3)$$

Where 6.85 × 10^{-7} µCi/µg is the specific activity of natural uranium [ORAUT 2025c].

If the ^{235}U result is greater than the ^{238}U result, including samples where ^{235}U is positive and ^{238}U is negative, HEU is indicated and the ^{235}U result is used to determine the total uranium intake. The assumed enrichment depends on the relative ^{235}U/^{238}U values and site information. Contact the PIDS for assistance.

Once all samples analyzed with ICP-MS have been converted to total uranium activity, they are assessed as usual and can be combined with other uranium sample results that are also in units of total activity.

9.2.2.3 In Vivo Counts

Chest counts are sometimes used for the measurement of uranium. Like some of the urine analytical methods, MDAs vary, and some isotopes might not be reported or have large MDAs. Total uranium activity is determined by the same methods described for in vitro methods. However, because photons are monitored rather than the alpha particles that can be detected with in vitro analysis, the relative detection capabilities of the various isotopes are different from those in urine samples. The ^{235}U isotope is the most easily detected in vivo and has often been reported. The MDAs of the other isotopes are much larger. Because of this, ^{235}U is typically the isotope of choice for determining the total uranium activity. If DU was the material of interest to the site, ^{238}U might be the reported isotope. In this latter case, gammas from the ^{234}Th progeny are measured and equilibrium is assumed.

9.3 ASSIGNMENT OF THORON AND RADON DOSE

For lung cancers, IREP requires ^{222}Rn exposures to be entered in units of working level months (WLMs). Radon-220 (also known as thoron) and ^{219}Rn (actinon) exposures are also frequently recorded in these units, but because the decay products have characteristics that are sufficiently different from ^{222}Rn, the exposure model is not applicable to them. In these circumstances, the reported ^{220}Rn and ^{219}Rn values must be converted to dose. DCAS-TIB-0011, *Dose Conversion Factors for Radon WLM* [NIOSH 2018b], contains conversion factors from WLMs to dose for ^{220}Rn and ^{219}Rn. Because the exposure model applies specifically to the lung for radon, it also contains conversion factors for all other organs for ^{222}Rn.

Note that ^{222}Rn breath measurements are used to monitor for intakes of ^{226}Ra rather than radon. See ORAUT-OTIB-0025, *Estimation of Radium-226 Activity in the Body from Breath Radon-222 Measurements*, for additional information [ORAUT 2005a].

9.4 ASSESSMENT OF MIXTURES OF RADIONUCLIDES

When dealing with mixtures of materials, such as those discussed in Section 3.3.3, all components must be considered and summed for the dose comparison when determining the absorption type that is favorable to the claimant.

When several nuclides might have been present at a site, monitoring and reporting was frequently limited to the one or two primary dose contributors. Dose from additional radionuclides is assigned based on ratios to these primary nuclides. Principal examples of these ratios include weapons-grade

plutonium, RU, and mixed fission and activation products. Site profiles contain details of the plutonium and uranium mixtures specific to the site, as well as other mixtures that are site-specific. ORAUT-OTIB-0054 [ORAUT 2015] provides guidance on the assignment of mixed fission and activation products, with site-specific application details in the site profile.

For a best estimate, review the nuclides that are included in a ratio method and do not assign any that were directly monitored because doing so would account for the same nuclide twice. For example, RU contains plutonium and some sites that handled RU also handled plutonium. A bioassay sample for plutonium would account for any plutonium dose regardless of whether it came from a uranium or plutonium mixture.

10.0 OTHER ISSUES

10.1 ASSESSING WOUNDS

Wounds are typically characterized by two or more compartments, with a fraction of the material absorbed almost immediately into the bloodstream and additional components with longer half-lives from material that remains at the wound site. Information on wound assessment is in ORAUT-OTIB-0022, *Guidance on Wound Modeling for Internal Dose Reconstruction* [ORAUT 2005b] and NCRP Report 156 [NCRP 2006]. IMBA contains the default models described in the NCRP report. Contact the PIDS for assistance with wound modeling. Note that wound models are not included in Web CAD so the doses *must* be assessed in IMBA.

10.2 CHELATION

Medical intervention in the form of a chelating agent is sometimes used to reduce the committed dose from intakes of certain transuranic elements (e.g., plutonium, americium, curium). The chelating agent binds to unbound material in the blood and forms a stable “chelate” complex that is then metabolized like the chelating agent, thus removing the transuranic element more quickly than its biological rate predicts. Chelation agents used for the treatment of occupational intakes of transuranic elements are the zinc and calcium salts of diethylene-triamine-pentaacetic acid (DTPA), better known as Zn-DTPA and Ca-DTPA. This enhanced excretion rate means that the standard urinary excretion model for the transuranic is not representative for some amount of time after chelation.

10.2.1 Assessment of Incident Intake Following Chelation

The default dose assessment approach for handling chelation-affected urine samples is the Jech method [Jech et al. 1972]. This method uses urinalysis data more than 100 days from the last chelation and the accepted ICRP biokinetic models to calculate the *effective* intake to assign dose. Contact the PIDS for guidance on assessments using fecal or chest count data.

Decisions are based on whether the bioassay data more than 100 days after the chelation are uncensored (i.e., a value is reported) or censored (no value is reported; “<” is used).

10.2.1.1 Uncensored Urine Result More than 100 Days after Last Chelation

- If a negative (<0) value is reported, contact the PIDS.
- Result >MDA:
  - Use Jech method to assess the effective intake using the bioassay result(s) after 100 days.

  - The subsequent bioassay samples might be >MDA or <MDA; follow typical program guidance for fitting bioassay data based on the specific situation.
  - Calculate the dose from the effective intake.
  - Assign dose as lognormal distribution with GSD of 3 (i.e., follow fitted dose assignment).

- Result ≤MDA:
  - Use Jech method to assess the effective intake using the bioassay result(s).
  - Enter first bioassay result >100 days after the last chelation as Real and if applicable, the immediate subsequent <MDA result as <LOD.
  - Calculate the dose from the effective intake.
  - Assign dose as lognormal distribution with GSD of 3 (i.e., follow fitted dose assignment).

10.2.1.2 Censored Urine Result More than 100 Days after Last Chelation

Contact the PIDS. Approach will be determined on a case-by-case basis.

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ATTACHMENT A
RETENTION OF MATERIAL IN THE LUNGS

LIST OF FIGURES

FIGURE **TITLE** **PAGE**

ATTACHMENT A
RETENTION OF MATERIAL IN THE LUNGS (continued)

All documentation for the plots in this attachment is in ORAUT [2025a].

As noted in Section 2.3, absorption type, also referred to as material type, describes the rate of absorption of deposited material in the respiratory tract into blood. Once in the blood the material is subsequently transferred to other organs and tissues, so it is not an indication of the total amount of time that the material is retained in or eliminated from the total body.

The very rapid clearance of Type F material from the lung after an acute inhalation intake is shown in Figure A-1. The material is transferred out of the lungs in less than a tenth of a day. Figures A-2, A-3, and A-4 show the clearance of Types M, S, and SS, respectively, after an acute inhalation intake. The same scale is used for all for comparison.

Figures A-5 through A-8 display the retention and clearance of the four types during and after a 365-day chronic inhalation intake of 1 Bq/d. The Type F material rapidly reaches equilibrium in the lungs, although a very small fraction of the intake is retained even during the intake period. The material is equally rapidly removed from the lungs once the intake stops. Note that the *y*-axis scale in Figure A-5 is 4 orders of magnitude smaller than that of Figures A-6 through A-8. Types M, S, and SS continue to increase in the lungs over a 365-day chronic intake, with the Type M activity reaching about half that of the less soluble types. After the cessation of intake, Type M is removed more rapidly but not immediately.

Chest counting is most often used for detecting the long-term buildup of strongly retained material in the lungs. Going to a longer intake of 1,000 days, it can be seen in Figure A-9 that the Type M material reaches an equilibrium state in the lungs shortly after about a year of chronic intake while the activity of Type S, plotted in Figure A-10, continues to increase throughout the period. Figure A-11 demonstrates that Type SS exhibits a similar pattern to Type S.

These plots demonstrate that chest counting cannot be used for assessing a Type F material because at any time during or after an intake little to no activity is expected to be present in the lungs. Chest counting can be used to detect Types M, S, or SS in the lungs but at any given time urine sampling might be more sensitive (i.e., able to detect smaller intakes) depending on the time since intake and the relative sensitivities of the two bioassay methods.

ATTACHMENT A
RETENTION OF MATERIAL IN THE LUNGS (continued)

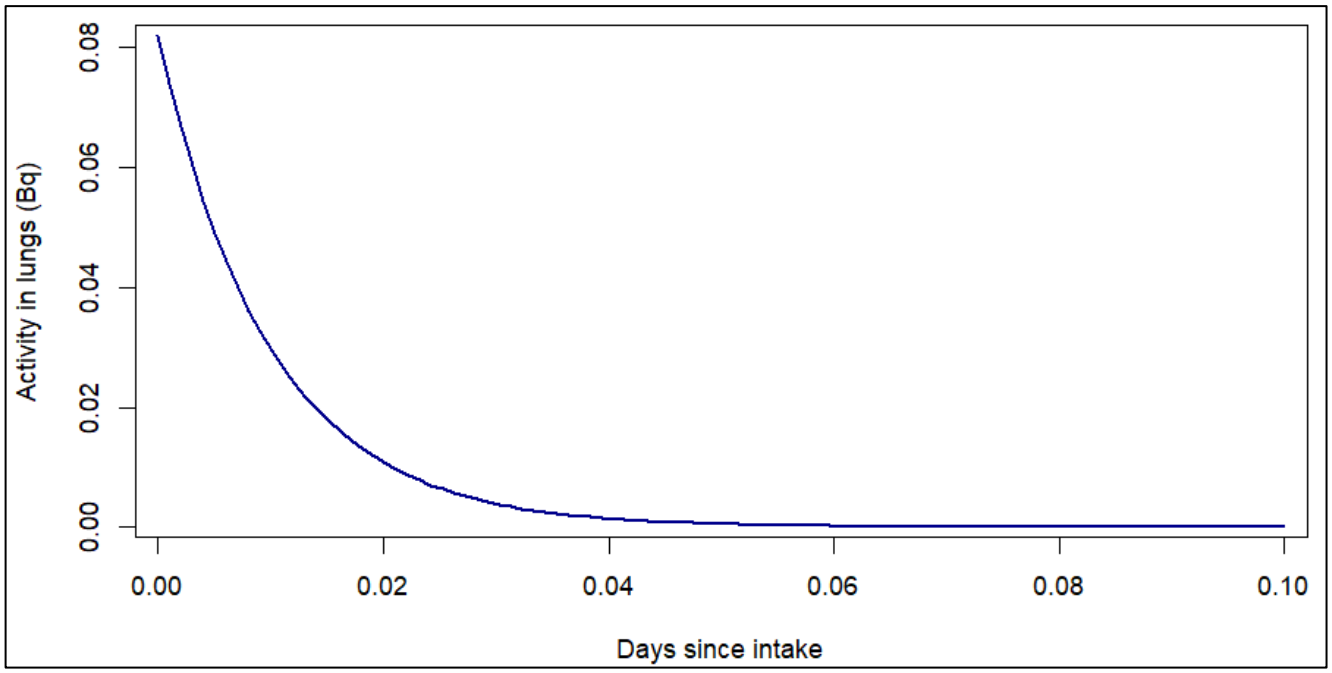

Figure A-1. Lung retention after an acute inhalation intake of 1 Bq of Type F material. Attachment E contains an extended description.

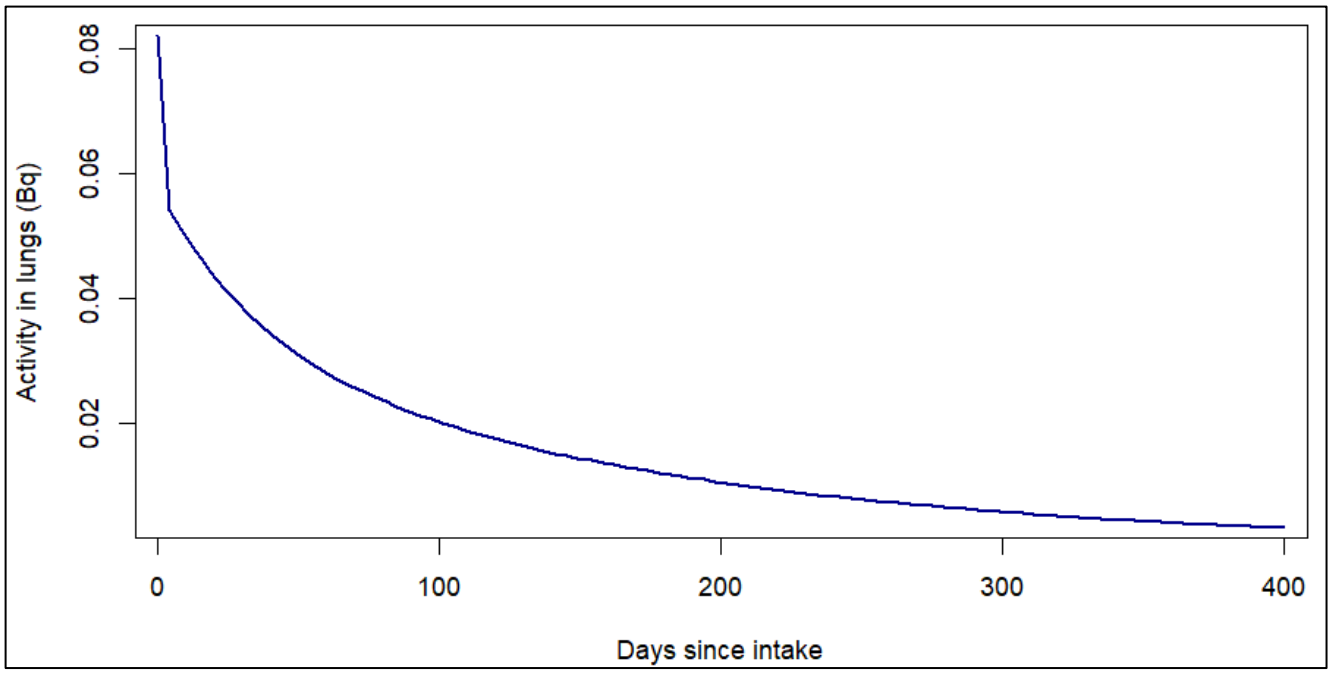

Figure A-2. Lung retention after an acute inhalation intake of 1 Bq of Type M material. Attachment E contains an extended description.

ATTACHMENT A
RETENTION OF MATERIAL IN THE LUNGS (continued)

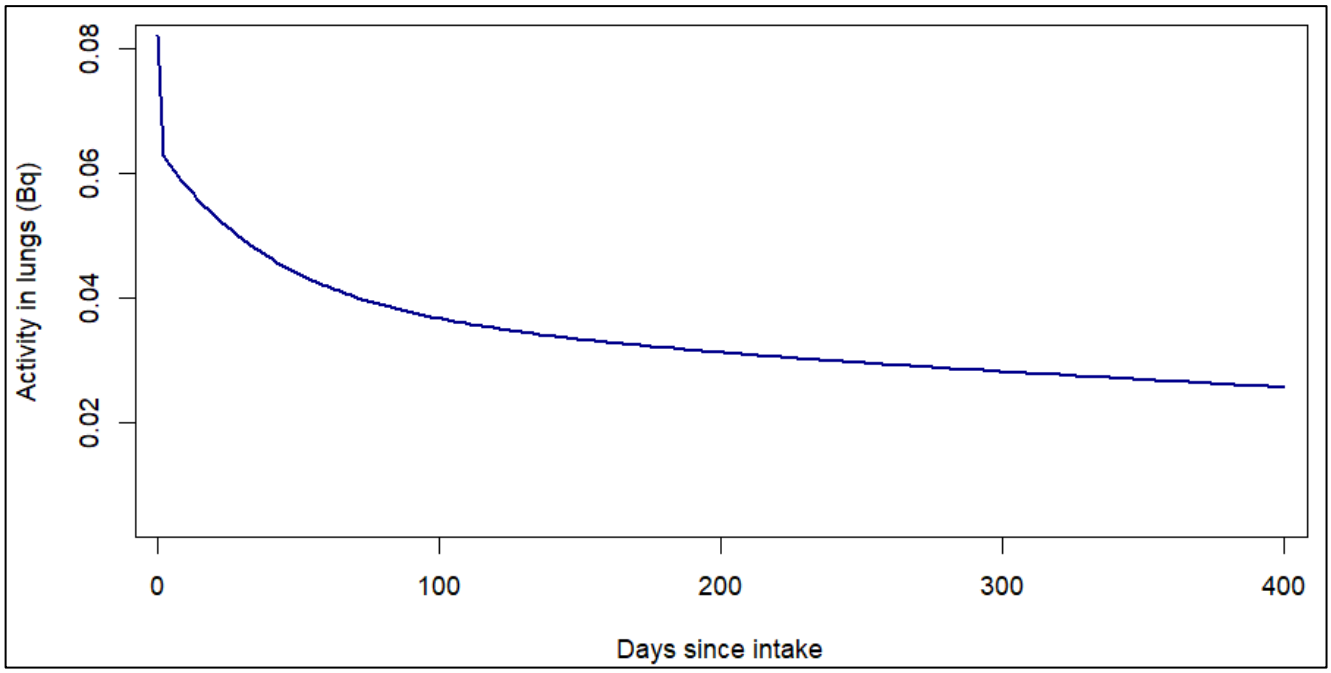

Figure A-3. Lung retention after an acute inhalation intake of 1 Bq of Type S material. Attachment E contains an extended description.

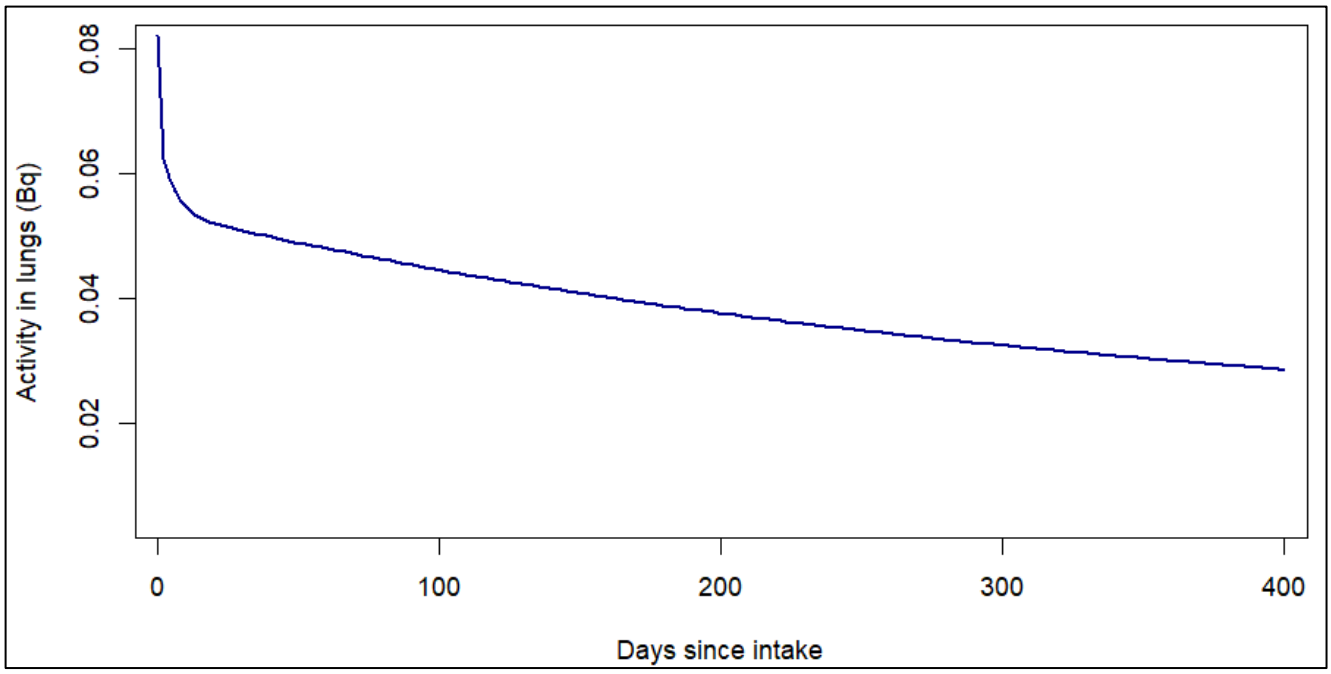

Figure A-4. Lung retention after an acute inhalation intake of 1 Bq of Type SS material. Attachment E contains an extended description.

ATTACHMENT A
RETENTION OF MATERIAL IN THE LUNGS (continued)

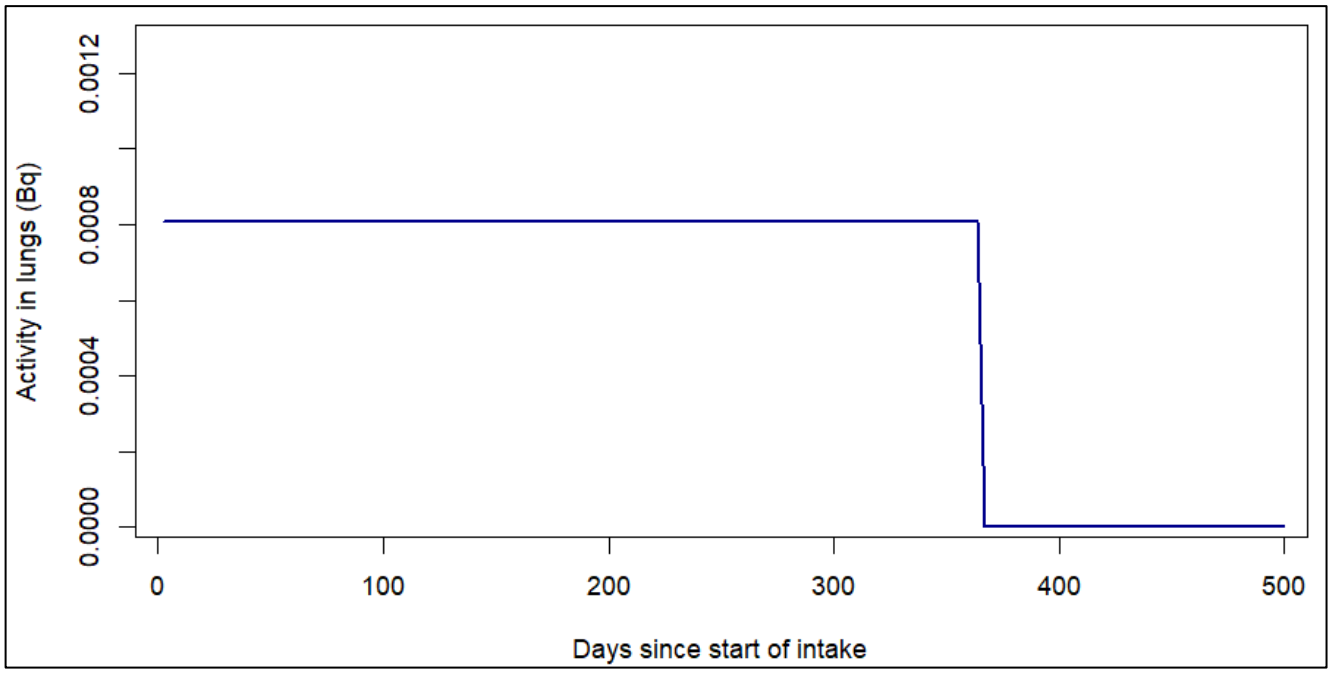

Figure A-5. Lung retention of a 365-day chronic inhalation intake of 1 Bq/d Type F material. Attachment E contains an extended description.

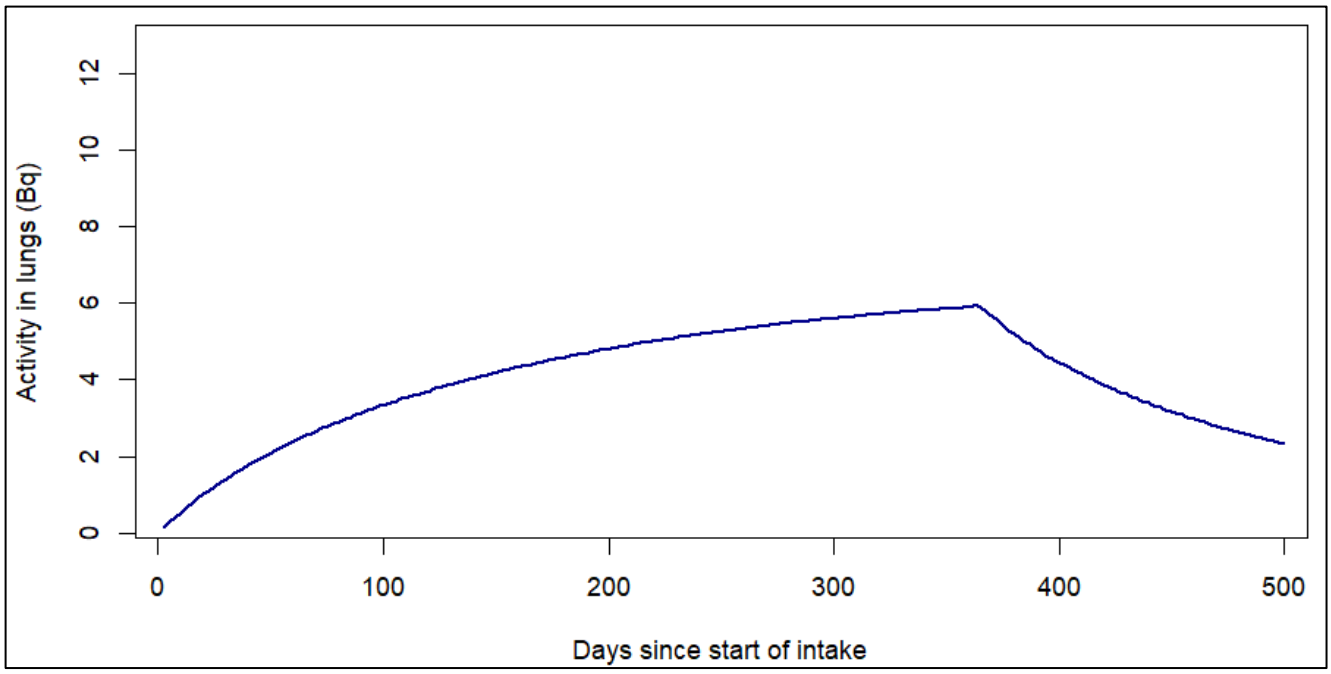

Figure A-6. Lung retention of a 365-day chronic inhalation intake of 1 Bq/d Type M material. Attachment E contains an extended description.

ATTACHMENT A
RETENTION OF MATERIAL IN THE LUNGS (continued)

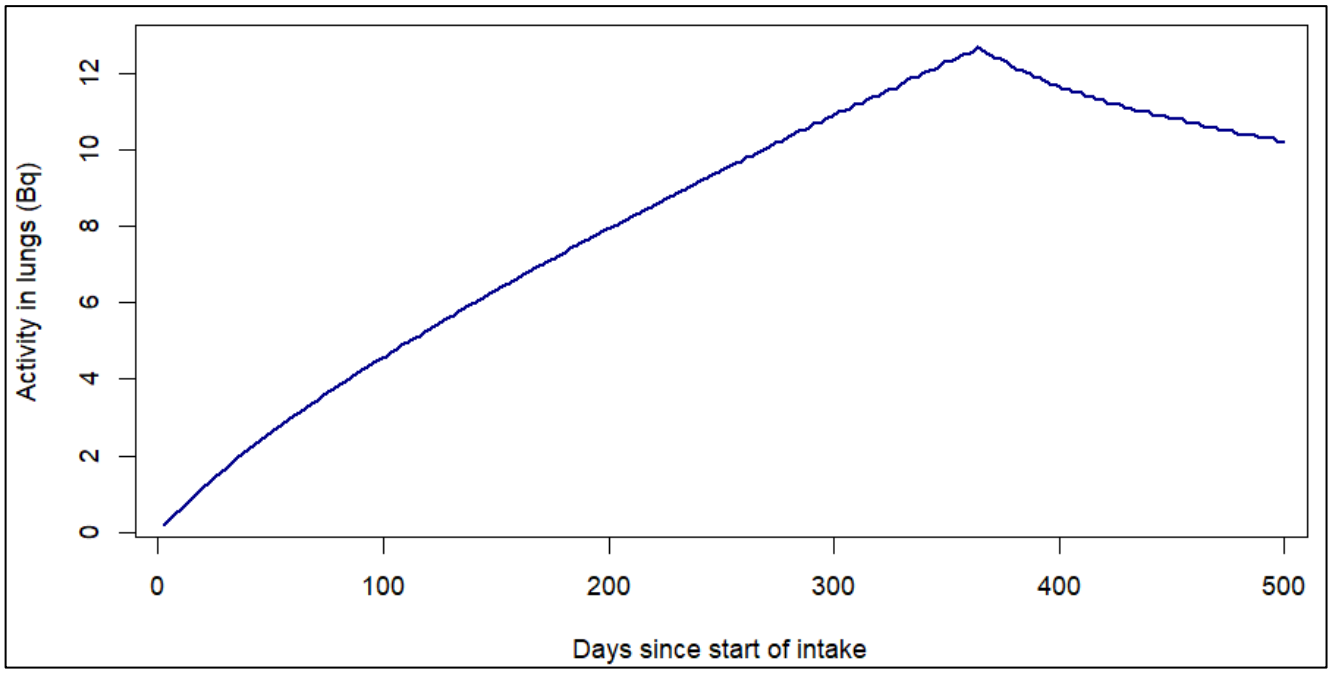

Figure A-7. Lung retention of a 365-day chronic inhalation intake of 1 Bq/d Type S material. Attachment E contains an extended description.

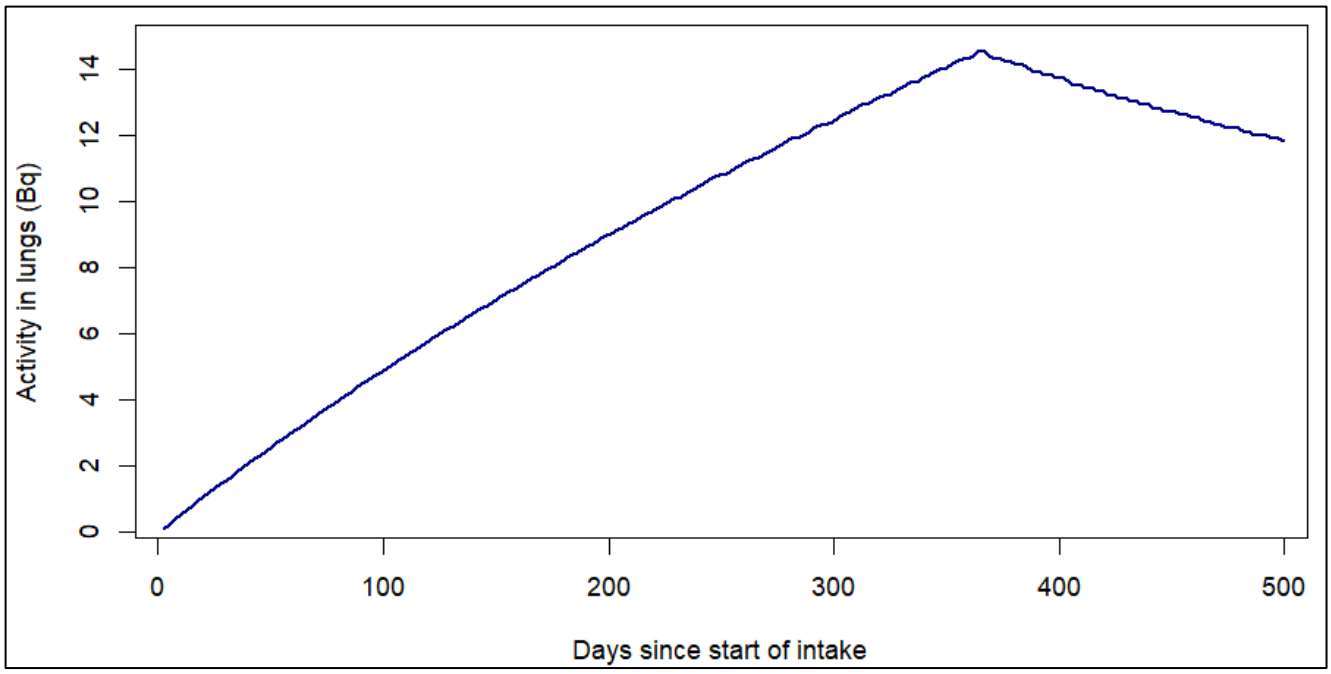

Figure A-8. Lung retention of a 365-day chronic inhalation intake of 1 Bq/d Type SS material. Attachment E contains an extended description.

ATTACHMENT A
RETENTION OF MATERIAL IN THE LUNGS (continued)

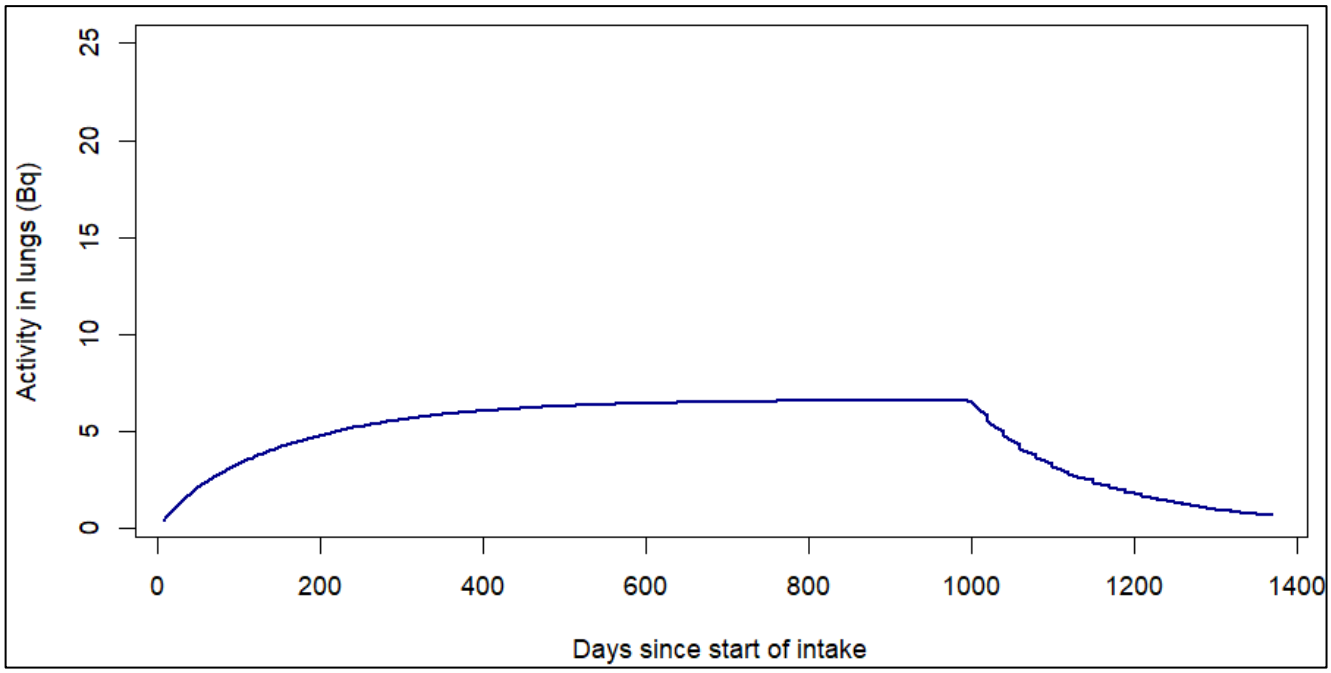

Figure A-9. Lung retention of a 1,000-day chronic inhalation intake of 1 Bq/d Type M material. Attachment E contains an extended description.

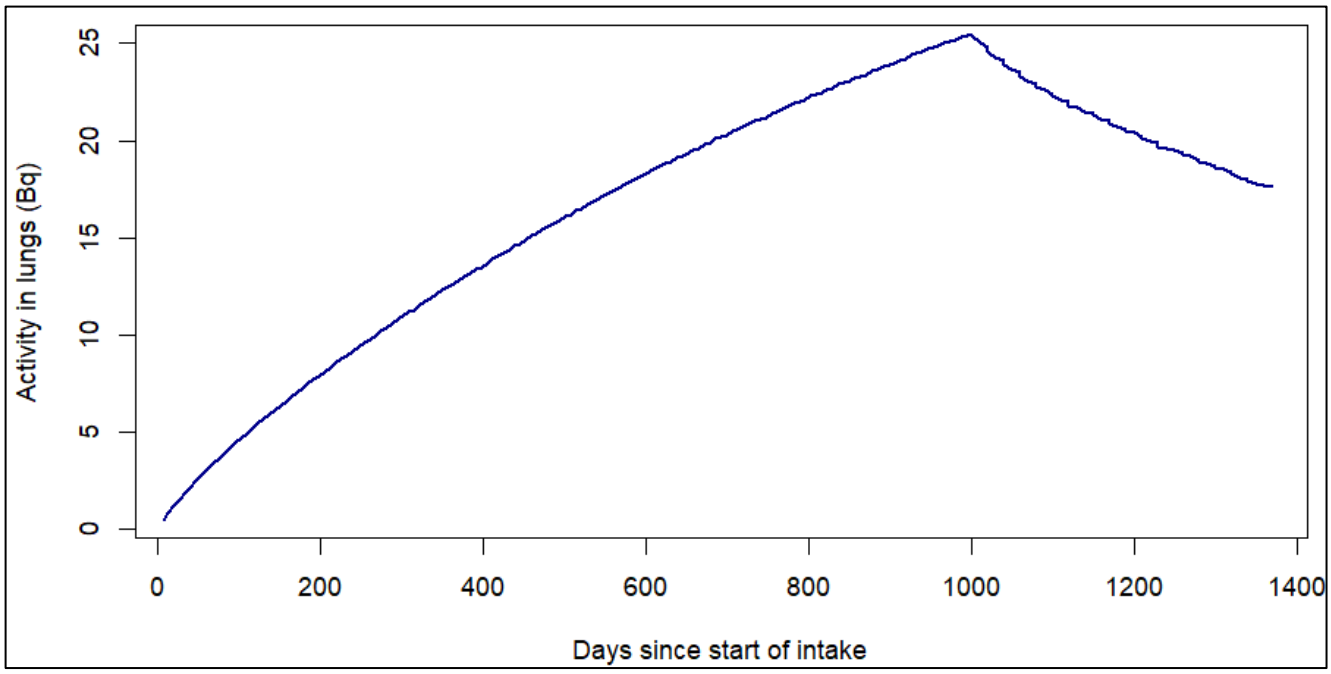

Figure A-10. Lung retention of a 1,000-day chronic inhalation intake of 1 Bq/d Type S material. Attachment E contains an extended description.

ATTACHMENT A
RETENTION OF MATERIAL IN THE LUNGS (continued)

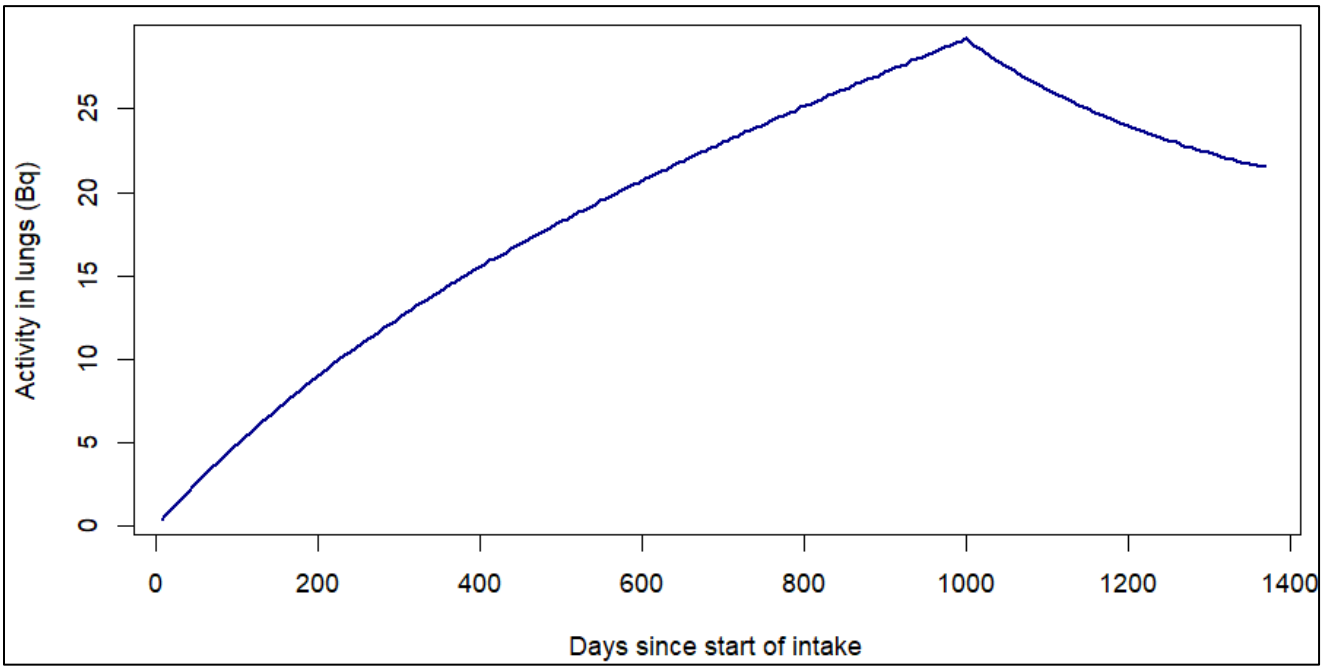

Figure A-11. Lung retention of a 1,000-day chronic inhalation intake of 1 Bq/d Type SS material. Attachment E contains an extended description.

ATTACHMENT B
ICRP PUBLICATION 68 DEFAULT MATERIAL TYPES

When modeling inhalation intakes, the rate at which material leaves the respiratory tract and is transported to the systemic organs is of importance. The ICRP [ICRP 1994b] uses material type (also called absorption type) to describe this rate. These designations apply *only* to material leaving the lung; they provide no indication of the total amount of time the material clears from the rest of the body. These types and their approximate half-times in the lung are:

- *V (vapor).* 100% is absorbed instantaneously.
- *F (fast).* 100% is absorbed from the lung with a half-time of 10 minutes.
- *M (moderate).* 10% at 10 minutes and 90% at 140 days.
- *S (slow).* 0.1% at 10 minutes, 99.9% at 7,000 days.

Table B-1 lists the ICRP default material types for all elements available for assessment in IMBA.

Table B-1. Publication 68 default material types.[a]

Atomic number	Element	Symbol	Material type(s)
1	Hydrogen	H	V
4	Beryllium	Be	M, S
6	Carbon	C	V
11	Sodium	Na	F
12	Magnesium	Mg	F, M
13	Aluminum	Al	F, M
14	Silicon	Si	F, M, S
15	Phosphorus	P	F, M
16	Sulphur	S	F, M, V
17	Chlorine	Cl	F, M
19	Potassium	K	F
20	Calcium	Ca	M
21	Scandium	Sc	S
22	Titanium	Ti	F, M, S
23	Vanadium	V	F, M
24	Chromium	Cr	F, M, S
25	Manganese	Mn	F, M
26	Iron	Fe	F, M
27	Cobalt	Co	M, S
28	Nickel	Ni	F, M
29	Copper	Cu	F, M, S
30	Zinc	Zn	S
31	Gallium	Ga	F, M
32	Germanium	Ge	F, M
33	Arsenic	As	M
34	Selenium	Se	F, M
35	Bromine	Br	F, M
37	Rubidium	Rb	F
38	Strontium	Sr	F, S
39	Yttrium	Y	M, S
40	Zirconium	Zr	F, M, S
41	Niobium	Nb	M, S
42	Molybdenum	Mo	F, S

ATTACHMENT B
ICRP PUBLICATION 68 DEFAULT MATERIAL TYPES (continued)

Atomic number	Element	Symbol	Material type(s)
43	Technetium	Tc	F, M
44	Ruthenium	Ru	F, M, S
45	Rhodium	Rh	F, M, S
46	Palladium	Pd	F, M, S
47	Silver	Ag	F, M, S
48	Cadmium	Cd	F, M, S
49	Indium	In	F, M
50	Tin	Sn	F, M
51	Antimony	Sb	F, M
52	Tellurium	Te	F, M
53	Iodine	I	F, V
55	Cesium	Cs	F
56	Barium	Ba	F
57	Lanthanum	La	F, M
58	Cerium	Ce	M, S
59	Praseodymium	Pr	M, S
60	Neodymium	Nd	M, S
61	Promethium	Pm	M, S
62	Samarium	Sm	M
63	Europium	Eu	M
64	Gadolinium	Gd	F, M
65	Terbium	Tb	M
66	Dysprosium	Dy	M
67	Holmium	Ho	M
68	Erbium	Er	M
69	Thulium	Tm	M
70	Ytterbium	Yb	M, S
71	Lutetium	Lu	M, S
72	Hafnium	Hf	F, M
73	Tantalum	Ta	M, S
74	Tungsten	W	F
75	Rhenium	Re	F, M
76	Osmium	Os	F, M, S
77	Iridium	Ir	F, M, S
78	Platinum	Pt	F
79	Gold	Au	F, M, S
80	Mercury	Hg	F, V
81	Thallium	Tl	F
82	Lead	Pb	F
83	Bismuth	Bi	F, M
84	Polonium	Po	F, M
85	Astatine	At	F, M
87	Francium	Fr	F
88	Radium	Ra	M
89	Actinium	Ac	F, M, S
90	Thorium	Th	M, S
91	Protactinium	Pa	M, S
92	Uranium	U	F, M, S
93	Neptunium	Np	M
94	Plutonium	Pu	M, S
95	Americium	Am	M

ATTACHMENT B
ICRP PUBLICATION 68 DEFAULT MATERIAL TYPES (continued)

Atomic number	Element	Symbol	Material type(s)
96	Curium	Cm	M
97	Berkelium	Bk	M
98	Californium	Cf	M
99	Einsteinium	Es	M
100	Fermium	Fm	M
101	Mendelevium	Md	M

a. Source: ICRP [1994b].

ATTACHMENT C
A METHOD FOR HANDLING SAMPLE RECOUNTS

Note: This attachment documents a method for handling recounts. It does not provide guidance on when or where it is to be applied. Contact the PIDS for assistance.

Question

If a sample is counted multiple times and only the results and their uncertainties (errors) are reported, how is it determined if the sample is positive?

Background

From ORAUT [2022]:

> *The general protocol in dose reconstruction is to use the MDA to decide if a result indicates the presence of net activity in the sample, i.e., to decide if the sample is positive:*
>
> - *If a result is > MDA then that result is used as given in the intake calculation.*
> - *If a result is ≤ MDA then the MDA/2 is used in the missed intake calculation.*

This general dose reconstruction guidance specifies how to determine if a result is positive. For this issue, additional questions include:

1. How to take a sample counted multiple times and come up with a summary result to compare to the MDA.
2. How to determine the MDA to compare to the summary result from item 1.

Weighted Mean Result (Item 1)

Recall that the sample is counted multiple times, and only the results and their uncertainties are reported, so the weighted mean of the results is used as the summary result to compare to the MDA (determined in the next section).

Suppose the sample is counted *n* times, the *n* results are:

$$x_1,\ x_2, \ldots\ x_n$$

and their corresponding 1σ uncertainties are:

$$\sigma_1,\ \sigma_2, \ldots\ \sigma_n$$

The weighted mean is [Cember and Johnson 2009, p. 518]:

$$\overline{x}_w = \frac{\sum_{i=1}^{n} w_i x_i}{\sum_{i=1}^{n} w_i} \qquad \text{(C-1)}$$

ATTACHMENT C
A METHOD FOR HANDLING SAMPLE RECOUNTS (continued)

where

$$w_i = \frac{1}{\sigma_i^2} \tag{C-2}$$

Note that Equation C-1 simplifies to the familiar arithmetic mean (or average) when the weights are all equal (i.e., all the uncertainties are the same).

To make a detection decision, the weighted mean of the results is the summary result to compare to the MDA (determined in the next section).

MDA (via Standard Error of Weighted Mean) (Item 2)
To determine the MDA to compare to the weighted mean from item 1, we must first look at the standard error of the weighted mean [Cember and Johnson 2009, p. 518]:

$$\sigma_w = \sqrt{\frac{1}{\sum_{i=1}^{n} w_i}} \tag{C-3}$$

where the weights are as described by Equation C-2. Note that Equation C-3 simplifies to the familiar standard error of the mean when the weights are all equal (i.e., all the uncertainties are the same).

Following the guidance in ORAUT [2022], when only uncertainty is given, MDA is 3.29 times the uncertainty for a single measurement. This attachment addresses a sample that is counted multiple times, where the standard error of the weighted mean was computed using all the measurements' uncertainties. Therefore:

$$MDA = 3.29\sigma_w \tag{C-4}$$

Answer: If a sample is counted multiple times and only the results and their uncertainties are reported:

1. If the weighted mean (Equation C-1) is greater than MDA (Equation C-4), then the sample is positive, and the weighted mean is used in the intake calculation.

2. If the weighted mean (Equation C-1) is less than or equal to MDA (Equation C-4), then the sample is not positive, and MDA/2 is used in the missed intake calculation.

Example
A worker has 5 counts of a single ^{239}Pu sample on 12/08/2005. Table C-1 below shows the 5 results and their uncertainties (specified as 1-sigma uncertainty). Weights, as defined in Equation C-2, are also included in the table.

ATTACHMENT C
A METHOD FOR HANDLING SAMPLE RECOUNTS (continued)

Table C-1. Worker ^{239}Pu sample on 12/08/2005.[a]

Result (dpm)	Uncertainty (dpm)	Weight
0.004779	0.002402	173322.1
0.006111	0.002513	158348.9
−0.000220	0.002584	149766.6
−0.001300	0.002308	187727.7
−0.003030	0.001753	325414.0

a. Sources: LLNS [1998–2008, p. 9]; ORAUT [2025a].

From Equation C-1, the weighted mean is:

$$\bar{x}_w = \frac{\sum_{i=1}^{n} w_i x_i}{\sum_{i=1}^{n} w_i} = 0.000536 \text{ dpm} \quad \text{(C-5)}$$

From Equation C-3, the standard error of the weighted mean is:

$$\sigma_w = \sqrt{\frac{1}{\sum_{i=1}^{n} w_i}} = 0.001003 \text{ dpm} \quad \text{(C-6)}$$

Using Equation C-4, the MDA is:

$$MDA = 3.29\sigma_w = 0.003299 \text{ dpm} \quad \text{(C-7)}$$

Using the logic from the “Answer” section:

- The weighted mean (0.000536 dpm) is less than the MDA (0.003299 dpm), so the sample is not positive, and MDA/2 (0.001649 dpm) would be used in the missed intake calculation.

ATTACHMENT D
RADIATION TYPES BY NUCLIDE FOR ENTRY INTO IREP

ORAUT [2006] contains a discussion about radiation types used in IREP and a general description of the assignment method. The list in ORAUT [2006] was expanded over time primarily using Radiological Toolbox [Eckerman and Sjoreen 2014] as a reference.

Table D-1. Radiation types by nuclide for entry into IREP.

Nuclide	Radiation type[a]
Ac-227	Alpha
Ac-228	Alpha
Ag-110m	Photons E >250 keV
Am-241	Alpha
Am-243	Alpha
As-74	Photons E >250 keV
As-76	Electrons E >15 keV
Au-194	Photons E <30 keV
Ba-133	Photons E >250 keV
Ba-140	Photons E >250 keV
Bk-249	Alpha
C-14	Electrons E >15 keV
Ca-45	Electrons E >15 keV
Ce-139	Photons E = 30–250 keV
Ce-141	Electrons E >15 keV
Ce-143	Electrons E >15 keV
Ce-144	Electrons E >15 keV
Cf-249	Alpha
Cf-252	Alpha
Cl-36	Electrons E >15 keV
Cm-242	Alpha
Cm-243	Alpha
Cm-244	Alpha
Co-57	Photons E = 30–250 keV
Co-58	Photons E >250 keV
Co-60	Photons E >250 keV
Cr-51	Photons E >250 keV
Cs-134	Electrons E >15 keV
Cs-137	Electrons E >15 keV
Eu-152	Photons E >250 keV
Eu-154	Electrons E >15 keV
Eu-155	Electrons E >15 keV
Eu-156	Electrons E >15 keV
Fe-55	Electrons E <15 keV
Fe-59	Electrons E >15 keV
H-3	Electrons E <15 keV
Hf-181	Electrons E >15 keV
I-125	Photons E <30 keV
I-129	Photons E = 30–250 keV
I-131	Electrons E >15 keV
I-133	Electrons E >15 keV
I-134	Photons E >250 keV
I-135	Photons E >250 keV
Ir-192	Photons E >250 keV
La-140	Photons E >250 keV
Lu-174	Photons E = 30–250 keV
Mn-54	Photons E >250 keV
Mn-56	Electrons E >15 keV
Mo-99	Electrons E >15 keV
Na-22	Photons E >250 keV
Na-24	Photons E >250 keV
Nb-94	Photons E >250 keV
Nb-95	Electrons E >15 keV
Ni-63	Electrons E >15 keV
Np-237	Alpha
Np-239	Alpha
P-32	Electrons E >15 keV
P-33	Electrons E >15 keV
Pa-231	Alpha
Pa-233	Alpha
Pa-234	Alpha
Pb-210	Alpha
Pm-147	Electrons E >15 keV
Po-208	Alpha
Po-209	Alpha
Po-210	Alpha
Pr-143	Photons E >250 keV
Pr-147	Electrons E >15 keV
Pu-236	Alpha
Pu-238	Alpha
Pu-239	Alpha
Pu-240	Alpha
Pu-241	Alpha
Pu-242	Alpha
Ra-220	Alpha
Ra-223	Alpha
Ra-224	Alpha
Ra-226	Alpha
Ra-228	Alpha
Rn-219	Alpha
Rn-220	Alpha
Rn-222 (lung)	Radon
Rn-222 (all but lung)	Alpha
Ru-103	Electrons E >15 keV
Ru-106	Electrons E >15 keV
S-35	Electrons E >15 keV
Sb-124	Photons E >250 keV
Sb-125	Photons E >250 keV
Sc-46	Photons E >250 keV
Sm-151	Electrons E >15 keV
Sn-113	Electrons E >15 keV
Sr-85	Photons E >250 keV
Sr-89	Electrons E >15 keV
Sr-90	Electrons E >15 keV
Sr-91	Electrons E >15 keV
Ta-182	Photons E >250 keV
Tb-160	Electrons E >15 keV
Tc-99	Electrons E >15 keV
Te-131	Photons E >250 keV
Te-131M	Electrons E >15 keV
Th-228	Alpha
Th-230	Alpha
Th-232	Alpha
Th-234	Alpha
Tl-201	Photons E = 30–250 keV
Tl-202	Photons E >250 keV
Tl-204	Electrons E >15 keV
Tm-170	Electrons E >15 keV
U-232	Alpha
U-234	Alpha
U-235	Alpha
U-236	Alpha
U-238	Alpha
U-239	Alpha
Y-88	Photons E >250 keV
Y-90	Electrons E >15 keV
Y-91	Electrons E >15 keV
Yb-169	Photons E >250 keV
Zn-65	Photons E >250 keV
Zr-95	Electrons E >15 keV

a. E – energy.

ATTACHMENT E
EXTENDED DESCRIPTIONS OF FIGURES

This attachment contains extended descriptions for figures and equations that exceed the character limit for inline alt text. These descriptions are provided to enhance accessibility for screen reader users.

Figure 2-1
Scatter plot with uranium fraction of intake in urine ranging from 1×10^{-7} to 1×10^{-2} on the *y*-axis and days after intake ranging from 0 to 3,500 on the *x*-axis. The Type F curve starts above the others at time 0 at close to 1×10^{-2}. It drops rapidly to about 5×10^{-4} over a few days, then drops less rapidly to about 5×10^{-6} at 500 days and finishes with a straight line to a value of about 1×10^{-6} at 3,500 days.

The Type M curve starts at about 1×10^{-3} and drops somewhat slowly to about 1×10^{-6} at 1,000 days, then intersects the Type F curve at about 100 and 750 days. It then continues in a straight line to a value of about 5×10^{-7} at 3,500 days.

The Type S curve initially drops much more rapidly than those for Types F and M, falling to 5×10^{-6} within the first few days. It then continues in an almost straight line to about 7×10^{-7} at 3,500 days. Starting at about 750 days it falls between the Types F and M curves.

Figure 3-1
Screen capture of IMBA popup window with table of uranium information. It has been recreated here as Table E-1.

Table E-1. Absorption types and f1 values for uranium.

Absorption type	f1	ICRP source	Chemical form
F	0.02	71	No data
M	0.02	71	Recommended default in the absence of specific information
S	0.002	71	No data
F	0.02	68	Most hexavalent compounds (UF_6, UO_2F_2 and $UO_2(NO_3)_2$)
M	0.02	68	Less soluble compounds (UO_3, UF_4, UCl_4 and most other hexavalent compounds)
S	0.002	68	Highly insoluble compounds (UO_2 and U_3O_8)
Ing	0.02	68	Unspecified compounds
Ing	0.002	68	Most tetravalent compounds (e.g., UO_2, U_3O_8, UF_4)

Figure 5-1
Scatter plot of urine sample results against days after the start of intake. The *y*-axis is a linear scale ranging from 1.6 to 3.0, and the *x*-axis goes from 160 to 1,030 days. There are a few dozen points on the plot randomly scattered from about 1.6 to 3.0. The data were fit by using the result at 800 days as indicated by a square for the result while the rest are circles. The line runs approximately through the middle of the results with about half falling above the line, half below, and two directly on it.

Figure A-1
Plot of activity in lungs after an acute intake of 1 Bq of Type F material as a function of time. The *y*-axis goes from 0 to 0.08 Bq, and the *x*-axis goes from 0 to 0.1 days. The line starts at a peak of 0.08 Bq at time 0 and drops in a curve toward *x*-axis. It approaches the *x*-axis asymptotically and is less than 1×10^{-4} Bq by 0.07 days.

Figure A-2
Plot of activity in lungs after an acute intake of 1 Bq of Type M material as a function of time. The *y*-axis goes from 0 to 0.08 Bq, and the *x*-axis goes from 0 to 400 days. The line starts at a peak of

ATTACHMENT E
EXTENDED DESCRIPTIONS OF FIGURES (continued)

0.082 Bq at time 0, drops very sharply to about 0.055 Bq within first few days, then continues to drop in long curve toward the *x*-axis. It appears to intersect the *x*-axis at close to 400 days.

Figure A-3
Plot of activity in lungs after an acute intake of 1 Bq of Type S material as a function of time. The *y*-axis goes from 0 to 0.08 Bq, and the *x*-axis goes from 0 to 400 days. The line starts at peak of 0.082 Bq at time 0, drops very sharply to about 0.06 Bq within first few days, then continues in very slow drop toward *x*-axis. At 400 days lung activity is about 0.027 Bq.

Figure A-4
Plot of activity in lungs after an acute intake of 1 Bq of Type SS material as a function of time. The *y*-axis goes from 0 to 0.08 Bq, and the *x*-axis goes from 0 to 400 days. The line starts at a peak of 0.082 Bq at time 0, drops very sharply to about 0.06 Bq within first few days, then continues in very slow drop toward *x*-axis. At 400 days lung activity is about 0.029 Bq.

Figure A-5
Plot of activity in lungs from a 1-Bq/d intake of Type F material for 365 days as a function of time. The *y*-axis goes from 0 to 1.2×10^{-3} Bq, and the *x*-axis goes from 0 to 500 days. The line starts at about 2 days with an activity value of about 8×10^{-4} Bq, then continues at this value in a straight line to 365 days, where it drops straight down to 0 Bq.

Figure A-6
Plot of activity in lungs from a 1-Bq/d intake of Type M material for 365 days as function of time. The *y*-axis goes from 0 to 12 Bq, and the *x*-axis goes from 0 to 500 days. The line starts in the lower lefthand corner of graph, slowly arcs up to about 6.4 Bq at 365 days, then drops off to about 2.6 Bq at 500 days.

Figure A-7
Plot of activity in lungs from a 1-Bq/d intake of Type S material for days as a function of time. The *y*-axis goes from 0 to 12 Bq, and the *x*-axis from 0 to 500 days. The line starts in the lower lefthand corner and rises in an almost straight line to 13 Bq at 365 days, then drops off to about 10 Bq at 500 days.

Figure A-8
Plot of activity in lungs from a 1-Bq/d intake of Type SS material for 365 days as function of time. This plot is very similar to the Type S plot. The *y*-axis goes from 0 to 14 Bq, and the x-axis goes from 0 to 500 days. The line starts in the lower lefthand corner and rises in almost straight line to 14.5 Bq at 365 days, then drops off to about 12 Bq at 500 days

Figure A-9
Plot of activity in lungs from a 1-Bq/d intake of Type M material for 1,000 days as a function of time. The *y*-axis goes from 0 to 25 Bq, and the *x*-axis goes from 1 to 1,400 days. The line starts in the lower lefthand corner, rises to a relatively flat line of about 7.5 Bq by about day 450 and continues to day 1,000, then drops almost to the *x*-axis by day 1,400.

Figure A-10
Plot of activity in lungs from a 1-Bq/d intake of Type S material for 1,000 days as a function of time. The *y*-axis goes from 0 to 25 Bq, and the *x*-axis goes from 1 to 1,400 days. The line starts in the lower lefthand corner and rises in a slight arch to 25.4 Bq at day 1,000, then drops in a slightly concave line to 17 Bq by day 1,400.

ATTACHMENT E
EXTENDED DESCRIPTIONS OF FIGURES (continued)

Figure A-11
Plot of activity in lungs from a 1-Bq/d intake of Type SS material for 1,000 days as a function of time. The *y*-axis goes from 0 to 25 Bq, and the *x*-axis goes from 1 to 1,400 days. The line starts in the lower lefthand corner and rises in a very slight arch to 29.2 Bq at day 1,000, then drops in a slightly concave line to 21.5 Bq by day 1,400.