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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.
04/20/2018	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.

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ACRONYMS AND ABBREVIATIONS

BB   bronchial region
Bq   becquerel

CAD   Chronic Annual Dose (tool)
CFR   Code of Federal Regulations

d   day
DCAS   Division of Compensation Analysis and Support
DL   decision level
DOE   U.S. Department of Energy
DOL   U.S. Department of Labor
dpm   disintegrations per minute

EEOICPA   Energy Employees Occupational Illness Compensation Program Act of 2000
ET1   extrathoracic region 1
ET2   extrathoracic region 2

g   gram
GI   gastrointestinal
GSD   geometric standard deviation

hr   hour

ICD-9   *The International Classification of Diseases, 9th Revision*
ICRP   International Commission on Radiological Protection
IMBA   Integrated Modules for Bioassay Assessment
ICP-MS   inductively coupled plasma–mass spectrometry
IREP   Interactive RadioEpidemiological Program

keV   kiloelectron-volt, 1,000 electron-volts

L   liter
LLI   lower large intestine
LN(ET)   extrathoracic lymph nodes
LN(TH)   thoracic lymph nodes
LOD   limit of detection

MDA   minimum detectable amount
MDD   minimum detectable dose
mL   milliliter
MPC   maximum permissible concentration

nCi   nanocurie
NCRP   National Council on Radiation Protection and Measurements
NIOSH   National Institute for Occupational Safety and Health

ORAU   Oak Ridge Associated Universities

PER   Program Evaluation Report
PID   Principal Internal Dosimetrist
POC   probability of causation

REF   radiation effectiveness factor

SI   small intestine
SRDB Ref ID Site Research Database Reference Identification (number)
SRS   Savannah River Site

TIB   technical information bulletin

U.S.C.   United States Code
ULI   upper large intestine

wk   week
WLM   working level month

µCi   microcurie
µg   microgram
µm   micrometer

µrad   microrad
§   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). TIBs may be used to assist NIOSH staff in the completion of individual dose reconstructions.

In this document, the word "facility" is used as a general term for an area, building, or group of buildings that served a specific purpose at a site. It does not necessarily connote an "atomic weapons employer facility" or a "Department of Energy (DOE) facility" as defined in the Energy Employees Occupational Illness Compensation Program Act of 2000 [42 U.S.C. § 7384l(5) and (12)].

1.1   PURPOSE

The purpose of this TIB is to provide information and guidance for reconstructing internal dose and to document the rationale for selection of certain default parameters.

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 energy employee. There may be approaches in this document that are not applicable or appropriate to a specific claim.

The terminology and methods in this TIB are applicable for reconstructing doses on the Project and are not necessarily reflective of standard internal dosimetry practices in an operational program, where the program is being conducted to meet regulatory standards. Quantities of interest are different, and methods in this document are intended to be favorable to the claimant when a parameter is unknown rather than using 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 to be found 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.

Section 2.0 defines terminology. Section 3.0 provides guidance for internal dose reconstruction. Section 4.0 provides a listing of additional documents containing internal dosimetry guidance and documentation, along with a description of the content. Attributions and annotations, indicated by bracketed callouts and used to identify the source, justification, or clarification of the associated information, are presented in Section 5.0. Attachment A lists radiation types by nuclide for entry into the Interactive RadioEpidemiological Program (IREP), Attachment B lists the International Commission on Radiological Protection (ICRP) Publication 68 (ICRP 1995a) material types for all elements of interest, and Attachment C addresses retention of material in the lungs.

2.0   TERMINOLOGY

2.1   SYSTEMIC VERSUS NONSYSTEMIC

"Systemic" organs and tissues are those to which radioactive material is transferred through blood circulation.

"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)], and bronchial region (BB) (ICRP 1994).

- 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 a weighted average of the LLI and ULI) (ICRP 1979).

2.2   METABOLIC VERSUS NONMETABOLIC ORGANS

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 particular 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 1994), 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 to be metabolic.

2.3   BIOASSAY

2.3.1   In Vitro Bioassay

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

2.3.1.1   Urinalysis

Urine sampling is the most common bioassay method across the complex, and was the first bioassay method implemented at most of the covered facilities. Samples can be 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. These consist of a single voiding of the bladder.

- <u>24-hour</u>. 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.

- <u>Simulated 24-hour</u>. 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.

- <u>Minimum volume</u>. A worker can 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.

2.3.1.2   Fecal Samples

Fecal analysis is a very sensitive procedure 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, assuming 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 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.

2.3.2   <u>In Vivo Bioassay</u>

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, 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.

2.3.2.1   Whole-Body Counts

A whole-body count measures high-energy gamma rays being emitted from anywhere in the body, although some counters are collimated such 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 particular counter, and the site-specific instrumentation settings, but is often around 100 keV.

2.3.2.2   Chest Counts

Chest counts are intended as a measure of material in the lungs but include any activity in the ribs. Low-energy gammas are of interest in these counts, and they are primarily used for the detection of long-term deposition 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 C contains additional discussion and graphs on the retention of all three material types in the lungs.

2.3.2.3   Organ Counts

Thyroid counts are frequently used to assess intakes of iodine, which concentrates in the thyroid. IMBA has an option for performing an assessment from thyroid measurements.

Some sites performed skull counts in order to determine how much activity was deposited in the bones. This information was then used to correct chest counts for activity deposited in the ribs rather than in the lungs. This count type is not common and is not used in the dose assessment.

2.3.2.4   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 an assessment but can be used to help determine if an intake has occurred.

2.4   POSITIVE AND NEGATIVE RESULTS

2.4.1   Definitions

2.4.1.1   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 (HPS 2011). The term "definitely present" used 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 that 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 in order to decide if analyte is present in that sample.

2.4.1.2   Minimum Detectable Amount

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.

2.4.1.3   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, i.e., 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 employee files. That is, results less than the reporting level are considered less-than values and are recorded as 0 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.

2.4.2 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 that value. In the early years of the weapons complex, most sites used the MDA to determine if activity had been detected. In more recent years, sites have used DL to make the decision about the presence of activity. Results between the DL and MDA are accounted for in the missed dose calculation so are not considered to be positive when performing a dose reconstruction.

It is also 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.

Many sites recorded the value of the MDA in the employee records for results less than the MDA. In such cases, a result recorded with a value of the MDA is considered to be negative. 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.

A sample-specific MDA takes precedence over the generic MDA in the site profile.

2.4.3 Occupational Uranium Intakes

Uranium is naturally occurring in the earth's crust and can be found in the urine and feces of individuals with no occupational exposure. This uranium background can vary greatly dependent on local geology, eating habits, and water source. While some sites have set levels below which they consider a result to be attributable to background, any result other than a baseline sample that is positive (as defined in Section 2.4.2) is assumed to be due to an occupational exposure for the purposes of dose reconstruction.

2.4.4 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 NCRP Report 94 (NCRP 1987) for the years when this fallout was most likely to produce interference with occupational whole-body count results. 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 results 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.

2.5 TYPES OF DOSES AND ASSOCIATED DISTRIBUTIONS

This section describes the different types of doses that can be assigned to an individual, which depend on several factors including the site of employment, available data, and potential for exposure. Appropriate distributions to be entered into IREP are discussed. Section 3.3 discusses the applicable conditions for assigning each type of dose and provides guidance on the calculations.

2.5.1 Philosophy of Internal Dose Assignment

The general assumption for internal dose reconstruction is that an individual was constantly exposed to at least some level of airborne radioactivity throughout employment. At a minimum, ambient

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

Year	Body burden (Bq)	Body burden (nCi)	Annual absorbed dose (µrad)
1953	10	0.27	40
1954	40	1.08	130
1955	80	2.16	260
1956	160	4.32	570
1957	190	5.13	660
1958	240	6.48	820
1959	300	8.1	1,000
1960	250	6.75	860
1961	170	4.59	590
1962	220	5.94	770
1963	400	10.8	1,400
1964	700	18.9	2,400
1965	590	15.9	2,000
1966	360	9.72	1,200
1967	210	5.67	730
1968	130	3.51	450
1969	100	2.7	360
1970	100	2.7	360
1971	100	2.7	360
1972	100	2.7	360
1973	100	2.7	340
1974	60	1.62	210
1975	40	1.08	140
1976	60	1.62	200
1977	40	1.08	140

a. Source: NCRP Report 94 (NCRP 1987, Table B5).

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. Note that if a fitted dose has been calculated for all periods, there is no missed dose and it is not calculated.

2.5.2 Missed Dose

Missed dose is the potential dose 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). Missed dose is assigned using actual worker in vivo or in vitro bioassay measurements that are less than the MDA and worker-specific employment information. 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 is not assigned to workers. However, for the EEOICPA program, this dose is included in the dose reconstruction assessment and is assigned as the missed dose to the worker. The "true" dose is assumed to fall between 0 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, in order to

standardize the missed dose calculation, the EEOICPA program imputes this data by using one-half the censoring value (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 0, a maximum based on results at the MDA, and a mode based on results at one-half the MDA.

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 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. 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.

2.5.3   Fitted Dose

Fitted dose is the term for doses that are calculated from positive bioassay results (although some <MDA results may also be used in the fitting of the intake), which typically confirms that an intake did occur. This 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.

2.5.4   Environmental Intakes

Internal radiation doses to some employees 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 employees, 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. These are assigned as a lognormal distribution with a minimum GSD of 3 to account for the uncertainty noted in Section 2.5.3.

2.5.5   Coworker Dose

Coworker exposure distributions are developed from available dosimetric data from DOE or atomic weapons employer sites. ORAUT-PLAN-0014, *Coworker Data Exposure Profile Development,* (ORAUT 2004b), provides a generic discussion on the development of these datasets. ORAUT-OTIB-0019, *Analysis of Coworker Bioassay Data for Internal Dose Assignment* (ORAUT 2005b), contains more specific information on the derivation of internal dosimetry data parameters. 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 provide assumptions for intakes based on coworker dosimetry analyses.

Typically, workers with a significant potential for intake are 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 GSD is determined in part by the fitting of the data, but a minimum of 3 is assigned even if the modeling yields a smaller value.

3.0 INTAKE AND DOSE ASSESSMENT GUIDANCE

3.1 SELECTION OF APPROPRIATE ORGANS FOR DOSE ESTIMATE

Dose must be calculated to the organ in which the cancer originated. ORAUT-OTIB-0005, *Internal Dosimetry Organ, External Dosimetry Organ, and IREP Model Selection by ICD-9 Code* (ORAUT 2012), correlates the codes in *The International Classification of Diseases, 9th Revision, Clinical Modification, Sixth Edition* (ICD-9; PMIC 2007) with the appropriate organs and tissues to be modeled in the Integrated Modules for Bioassay Assessment (IMBA) computer program. For claims where the organ of cancer origination is not included in IMBA, 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 in IMBA that are not part of the ICRP (1998) metabolic model for the particular radionuclide.

In practice, dose reconstructors typically do not need to make this determination. The Chronic Annual Dose (CAD) tool, also referred to as Web CAD, selects the highest nonmetabolic organ for the selected ICD-9 code when appropriate by calculating the doses to each nonmetabolic organ and assigning the largest. The following is primarily for informational purposes.

Organs that do not concentrate a radionuclide receive photon exposure because of their proximity to the concentrating organs. The newer ICRP biokinetic models consider exposure from beta and alpha radiation to these other organs by defining them as a "soft tissue" compartment and describing uptake and clearance rates for this compartment. Many of these nonmetabolic organ doses are calculated using these techniques. Because these organs are all considered soft tissue, and are therefore all similarly exposed, all of the doses are relatively equal. This implies that choosing the largest of these doses is favorable to claimants. However, it is possible for one of the organ doses to be much higher than the others due to proximity to a concentrating organ that is emitting photon radiation. In this instance, the location of the cancer must be evaluated to ensure the estimate is not unrealistically large. If it is, the next largest organ dose should be used. For example, the thymus is relatively close to the thyroid so would not be included when determining the highest nonmetabolic organ for an intake of ^{131}I.

IMBA can be used to determine which organs are not included in the metabolic model. Dose reconstructors should:

1. Select the appropriate element (any isotope works) and load the ICRP defaults using the button on the toolbar.

2. Click **Biokinetics** (near the Close button).

3. Select **Load ICRP Defaults** in the window that pops up.

Several organ names will be highlighted in blue or purple at the top of the window. These are the modeled organs for the element (organs that are specifically named in the model as concentrating the radionuclide). Therefore, the remaining organs, with the exceptions noted in Section 2.2, are not part of the metabolic model and the largest dose among these organs (i.e., the highest nonmetabolic organ) is used for assigning doses to organs that are not included in the IMBA calculations.

3.2 SELECTION OF PARAMETER VALUES

Several parameters must be determined or assumed when assessing an internal dose. Specific knowledge of an individual claim or site takes precedence (see Section 1.2). However, when values

are unknown, default values should be applied. This section provides guidance on parameter selection.

3.2.1   Intake Mode

There are five intake routes included in IMBA:

- Inhalation is the most common route for workplace intakes.

- Ingestion can be associated with an inhalation intake. When an inhalation intake is assigned based on air monitoring rather than bioassay data, an additional ingestion component must 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 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. ORAUT-OTIB-0022, *Guidance on Wound Modeling for Internal Dose Reconstruction* (ORAUT 2005a), discusses modeling these intakes. Additional information can be found in National Council on Radiation Protection and Measurements (NCRP) Report 156 (NCRP 2006). IMBA contains the default models described in the NCRP report. Contact the Principal Internal Dosimetrist (PID) for assistance with wound modeling.

- Vapor is a specific instance of an inhalation intake; 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.

3.2.2   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 (ICRP 1994). 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.

3.2.3   Material Type

3.2.3.1   Standard Types

Material type describes the rate of absorption of deposited material in the respiratory tract into blood. ICRP Publication 66 describes three types: F (fast solubilization), M (moderate solubilization), and S (slow solubilization) (ICRP 1994). 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 1995a) are used for this Project because they address worker intakes rather than those of members of the public. These recommendations can be found in IMBA. Figure 3-1 shows an

example from the IMBA ICRP Defaults selection menu. If material types at a particular site are known, they are documented in the site profile and should be applied under the specifications in that document. Attachment C contains additional information about material types.

For the majority of claims, it is likely that the material type is unknown or 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 1995a) should be made, and the type that results in the largest dose to the organ of interest should be selected. 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 ^{241}Am in a plutonium matrix. Americium is assigned only 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
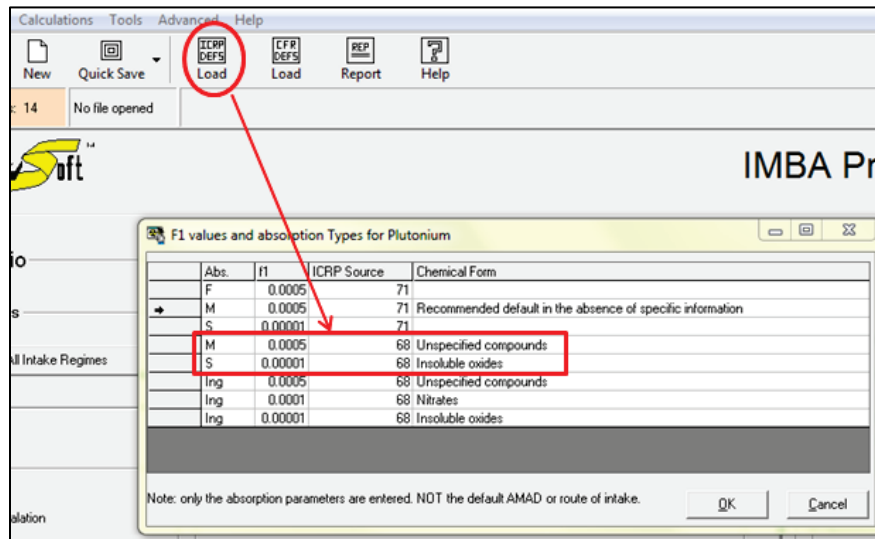
Figure 3-1. Using IMBA to determine Publication 68 default material types.

(ICRP 1995b) type S is selected in IMBA or Web CAD (or other assessment method that might be used) for ^{241}Am when in a type S plutonium matrix.

3.2.3.2   Additional Plutonium Types

When assessing ^{239}Pu, type super S could need to be considered in addition to types M and S. This type potentially applies to all methods of assigning intakes including coworker doses and efficiency methods. ORAUT-OTIB-0049, *Estimating Doses for Plutonium Strongly Retained in the Lung* (ORAUT 2010), discusses conditions under which this type applies and appropriate assessment methods. Associated ^{241}Am is also assessed as type super S. Plutonium-238 also has an alternate material type that is applied at specific sites under certain conditions. Details of this model and its applicability are documented in DCAS-RPT-005, *Alternative Dissolution Models for Insoluble Pu-238* (NIOSH 2016).

3.2.3.3   Intakes of Mixtures

When a mixture of nuclides is bound in a single matrix, the contaminants 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, the closest solubility should be selected. Examples of these mixtures include thorium and recycled uranium. Table 3-1

provides guidance on material types for several elements that may be included as a recycled uranium contaminant. Thorium mixtures are similarly assigned, matching the contaminant radionuclide types to the dominant thorium as closely as possible. See Section 3.7.4 for additional discussion on mixtures.

3.2.3.4   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. For example, Figure 3-1 includes the plutonium f1 values, while Figure 3-2 displays the uranium values. Table 3-2 contains a summary of selected f1 values for a recycled uranium mixture.

Table 3-1. Selected absorption types for recycled uranium 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

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 3-2. IMBA window showing uranium default material types and f1 values.

Table 3-2. Selected f1 values for recycled uranium 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

3.2.4   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 probability of causation (POC) calculation. The selected REF depends on the type of radiation, of which there are six:

- 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, Apostoaei, and Hoffman 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 A 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.

3.2.5   Normalization of Bioassay Data

The regulations for this Project require the use of ICRP models for calculating internal doses and specifies the use of certain publications (42 CFR Part 82). The biokinetic models and subsequent dose coefficients in those publications are based on male physiology and anatomy. Therefore, when urinalysis results are reported in units other than per day, the results are normalized to 24 hours using a conversion factor of 1.4 L/d for all individuals, male or female. For fecal samples, the default mass is 135 g/d.

Note that these values are to be used only for instances where a 24-hour sample has not been collected. If a 24-hour sample is indicated but the volume is not 1.4 L (or 135 g for fecal samples), do not adjust the result. Results must be normalized before IMBA entry. This applies to all data types (see Section 3.2.6.1).

If a sample volume is reported for a known 24-hour collection period and the results are reported per liter, it is appropriate to use the reported sample volume for the adjustment. If a sample volume is reported but there is no indication of the sample collection period and the site profile contains no additional information, 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.

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.

3.2.6 IMBA-Specific Parameters and Information

Additional information about IMBA and how it functions can be found in the IMBA documentation (under the Help menu in the program, in Documentation/Main documentation), in the IMBA procedure (ORAUT 2003b), and on the Project network on the obj3_dr (Q:) drive in the DR Information\Internal Dosimetry\IMBA folder.

3.2.6.1 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 2.4.1.3), as described in Section 3.4 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. These results are displayed in brown on the graph.

- Excluded. IMBA ignores all results with this Data Type when calculating the intake. Excluded results are displayed in red on the graph.

3.2.6.2 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.

IMBA allows the user to generate the measurement error by selecting the type of error relative to the measured result (uniform absolute, uniform relative, and square root) and a factor k by which it is multiplied. ORAUT-PROC-0002, *Use of Integrated Modules for Bioassay Analysis (IMBA)* (ORAUT

2003b), contains more details on the use of this function. Values entered as <LOD should have an error equal to the LOD (this is equivalent to k = 1 if using the Uniform Relative option).

For positive results, use the 1-sigma error associated with the result 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 is arbitrary; the same intake is calculated from any factor, given the same k for all results.

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 value.

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). Contact the PID for assistance if a reasonable fit cannot be obtained.

Note that changing the weighting factor (i.e., the Measurement Error) on the results simply moves the data fit up and down, it does not change the shape of the curve.

3.2.6.3   Error Distribution

"Normal" is typically selected for the distribution. Bioassay analyses tend to have one primary source of error that drives the uncertainty of the analysis. This uncertainty tends to be random about a mean value. As such, a normal distribution would be the most appropriate type of error distribution and is the assumed distribution type for this program.

3.2.6.4   IMBA Limitations

IMBA is known to calculate inaccurate organ doses for some radionuclide and organ combinations (Table 3-3). There are two categories of problems: shared versus independent kinetics and very short-lived radionuclides. In the case of the former, 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 are assumed to follow the behavior of the parent. 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.

Table 3-3. Nuclides for which IMBA calculates incorrect organ doses.

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

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 3-3, 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 a given radionuclide).

Contact the PID when a best estimate is needed for time frames that cannot be calculated with Web CAD. When sending a claim to the PID for a best estimate, be sure to include:

- 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.

3.3 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.

3.3.1 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
- External dose.

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 an individual'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, the assignment of environmental intakes only is appropriate. See Section 4.7 for additional information.

Note: Site-specific information typically takes the form of intakes based on air monitoring or source term information and is documented in the site profile.

3.3.2 Missed Dose Determination

A constant, chronic intake is assumed throughout the possible exposure period(s) unless there are changes in the MDA as noted below. The date of the last sample result that is <u>less than the MDA</u> 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. Note that if a fitted dose has been calculated for all periods, there is no missed dose and it is not calculated.

If the MDA changes during the intake period, the following must be considered in determining the chronic intake (note that "negative" here refers to a result <MDA):

- If the MDA decreases over time and the radionuclide/absorption type reaches equilibrium slowly in the compartment of interest (e.g., type M or S plutonium in urine, any type S material in the lung), perform the fit using the date of the last negative sample and half of the associated MDA and assume a single chronic intake for the entire potential exposure period. Only the lowest MDAs need to be considered in this scenario because any assessment of early values results in the overestimation of the later, smaller MDA values.

- If the MDA decreases over time for radionuclide/absorption types that reach equilibrium rapidly or if the MDA increases over time for any radionuclide/absorption type combination, use IMBA to determine chronic intakes applicable to each period. To do this:

  – In IMBA, set the number of intakes to the number of periods of different MDAs in which the worker has bioassay results (i.e., 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).

  – The first chronic intake period begins on the day the exposure began and continues to the date of the final sample with a given MDA.

  – 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.

  – Perform the fit assigning half of the associated MDA to the date of the last sample in each period.

- When the MDA oscillates, usually due to samples with individually reported 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 values would be suitable. Contact the PID for assistance if necessary.

Calculate the annual organ doses from the intakes of each radionuclide and enter the doses into IREP using a triangular distribution and:

1. Set the lower bound (Parameter 1) to 0.
2. Set the mode (Parameter 2) to the annual organ doses as calculated above.
3. Set the upper bound (Parameter 3) to 2 times the mode.

3.3.3   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. 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 recycled uranium are not typically monitored directly but can be assessed based on a ratio to the calculated uranium intake.

2. Coworker data. Assignment of coworker doses is discussed in Section 3.3.5.

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

4. Missed dose. Extension of the missed dose intake rate 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.

3.3.4   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 and 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, an individual 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.

An individual's bioassay data always take precedence over other data (e.g., coworker or site default values), unless the bioassay has been shown to be flawed or not representative of the individual'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. Types F and M are not retained for significant periods in the lungs and the rules for short-retained radionuclides must be followed.

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.

- Note that for very short-retained materials, notably ^{3}H and ^{131}I, this period is shorter. Guidance on ^{3}H assessment is contained in ORAUT-OTIB-0011, *Tritium Calculated and Missed Dose Estimates* (ORAUT 2004c). In general, ^{3}H 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 ^{3}H before the implementation of a bioassay program.

- 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 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.

3.3.5   Coworker Data

Coworker dose is applied as a best estimate for individuals with a potential for intakes of radioactive material but 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 coworker analyses 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, *Assignment of Environmental Internal Doses for Employees Not Exposed to Airborne Radionuclides in the Workplace* (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.

3.3.6 Example Assignment of Coworker 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
The MDAs remain constant over the employment period
Plutonium-239 urine samples on 05/4/1960, 12/11/1963, 11/17/1980
Strontium-90 urine samples on 03/12/1965, 09/18/1965, 08/01/1966, 01/05/1978, 07/12/1978, 04/30/1979

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:

   a. Assign coworker dose if it is available.

   b. If coworker data are not available but the site profile contains default intake values for individuals who were potentially exposed, assign the site profile values.

   c. If no other information is available, extend the missed dose through this period.

Strontium-90 dose calculation:

1. Assign coworker data or site default values from 03/01/1957 to 03/12/1964.

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

3. Assign coworker data or site default values from 08/02/1966 to 01/05/1977.

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

5. Assign coworker data or site default values from 05/01/1979 to 07/12/1989.

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

3.3.7 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 bioassay result is 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 coworker 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}.

3.4   FITTING POSITIVE BIOASSAY RESULTS

This section describes the process for fitting results to known intakes (i.e., 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 2003b).

3.4.1   <u>General Philosophy</u>

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 3.5 for discussion), no further fitting should be tried.

In general, the overall pattern of the data should be fit rather than each individual result. It is <u>not</u> 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.

3.4.2   <u>General Guidelines</u>

Assessment of positive bioassay results is subjective in the absence of known intakes, so the following guidelines are provided:

- Use all positive bioassay results, starting with the first positive value.

- For results that are not positive:

    – Enter the MDA value for both the Measurement Result and Measurement Error.

    – Include the first negative result after each set of positive results and set Data Type to **<LOD**.

    – If there are several consecutive positive results, include no more than two of the immediately following consecutive negative results. For fewer than five consecutive positive results, include only the first immediately following 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 of 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, use the 1-sigma error associated with the result when available. If there are no reported errors with the results, see the discussion in Section 3.2.6.2 for guidance.

- 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 3.3.2.

- 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 not be varied if there are only a few results for each intake unless projections from the intakes are inconsistent with later data (e.g., several negative results are predicted to have had detectable levels of activity). As discussed in Section 3.4.1, 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 <u>dose</u> 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).

- If the majority of 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 3.2.6.2) also applies here.

3.5   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, accurate 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.

3.5.1   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 individuals. Individual overestimates can also be made using individual-specific information.

3.5.1.1   Generic Overestimates

Several methods have been developed and documented in TIBs. Summaries and general applicability of these are described in Section 4.0; specific details are in each document.

Most of the overestimating methods are applicable to individuals with no positive bioassay results. However, this can be extended to individuals 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).

3.5.1.2   Individual-Specific Overestimates

Overestimates can sometimes be applied to individuals 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 by this ratio, then run the Intakes-to-Bioassay calculation to demonstrate that all bioassay results have been overpredicted.

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 of the results associated with the intake.

The constant distribution is assigned in IREP because this is an upper bound of the dose (NIOSH 2002).

3.5.2 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, missed dose only, or the underprediction of all or most positive bioassay results. 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 successful 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 complex. In such instances, a missed dose calculation alone might be adequate for determining compensability.

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

3.5.3 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 Energy Employees Occupational Illness Compensation Program Act of 2000 (EEOICPA). It can include some parameter values that are under- or overestimated if the outcome is consistent with those assumptions. That is, the POC is greater than 52% when underestimating parameters have been included or less than 45% where overestimates were used.

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.

3.5.3.1 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 Section 2.5.1 and the final bullet of Section 3.4.2. In such instances, there is no missed dose because all of 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 3.3.2. 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 3.4. 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.

3.5.3.2   Multiple Cancers

Consistent assumptions must be made for all cancers when performing a best estimate. For example, the material type for a given intake of a radionuclide must be the same for all organs. Note that this does not preclude the assignment of multiple material types for a given nuclide. This can happen when one type is maximizing for a fitted dose while a different type is more favorable to the claimant for the missed dose. However, 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.

Selection of the material type is based on that which yields the largest overall probability of causation.

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.

3.6   WORKING WITH MULTIPLE BIOASSAY TYPES

3.6.1   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 3.7.1.2.1 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.

3.6.2   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.

3.7   SPECIFIC ISSUES

3.7.1   Plutonium 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 of 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.

3.7.1.1   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 3.7.1.3 for guidance on using IMBA for this assessment.

3.7.1.2   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.

3.7.1.2.1   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. Although unlikely, a prediction from this intake to the urine sample results is then necessary to ensure that they are not overpredicted.

3.7.1.2.2   Age of Material

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, 10-year-old plutonium should be assumed. These assumptions can be used for either urinalysis or chest counting.

3.7.1.3   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 taken into account.

Initial set-up steps:

1. Open new session of IMBA.

2. In the Indicator Radionuclide box select **^{241}Am**.

3. In the same box, change the Number of Associated Radionuclides to **1**.

4. In the Associated Radionuclides box, select **^{241}Pu**.

5. In the same box, set the Abundance for the associated radionuclide.

    a. This value is 100 times the ratio of activity of ^{241}Pu to activity of ^{241}Am in the mixture.

   b. For example, for the Savannah River Site (SRS) 10-year-aged 12% mixture, this value is **4910**.

6. 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**.

7. 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.

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

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

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

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

To project ^{241}Am in lung count at any time after intake with a known initial intake of ^{241}Am and ^{241}Pu:

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.

3.7.2 Methods for Assessing Isotopic Uranium Results

Note: Isotopic results are from samples that were analyzed using alpha spectroscopy with individual results reported for ^{234}U, ^{235}U, ^{238}U, and sometimes ^{236}U. In the case of inductively coupled plasma–mass spectrometry (ICP-MS), typically only ^{235}U and ^{238}U are reported.

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 ICP-MS, each isotope is measured 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 recycled uranium mixtures. On the other hand, for a best estimate it is not appropriate to assess each component individually and sum the results unless all isotopes are >MDA because the isotopes are not present in equal concentrations for any mixture. For example, ^{235}U contributes only 2% of the total natural uranium activity so assuming it was present at the same activity as ^{234}U and ^{238}U (which would be the case if all isotopes were reported as <MDA and the MDA was used for each) would yield an overestimate, as noted below in Section 3.7.2.1.2.

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

3.7.2.1 Alpha Spectroscopy

The general rule of thumb for all claims is to start with the isotope with the highest activity. For depleted uranium this is ^{238}U; for all others it is ^{234}U.

3.7.2.1.1 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.

3.7.2.1.2 Overestimate

For an overestimate, the activity from all three isotopes can be summed and assessed as ^{234}U. For isotopes <MDA, add the MDA value (all three MDAs can be added 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.

3.7.2.1.3 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, each isotopic result needs to be adjusted to a total uranium activity value:

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

    a. Sum the individual results.

    b. 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 48.86% ^{234}U, 48.86% ^{238}U, and 2.28% ^{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:

$$\frac{(0.5+0.5)\,\text{dpm}}{0.486+0.486} = 1.029\,\text{dpm} \tag{3-1}$$

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.

4. If no isotopes >MDA, this calculation is the same as that in method 3 but is based on a single MDA. For depleted uranium, use the ^{238}U MDA and fraction; for all other enrichments use ^{234}U values.

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 depleted uranium, 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.

3.7.2.2 Inductively Coupled Plasma-Mass Spectrometry

This analytical technique measures the mass of a nuclide rather than activity. This is a relatively new technique that is inexpensive and quick for analyzing uranium so has been implemented at some uranium facilities in the last 20 or so years. Only ^{235}U and ^{238}U are typically reported because ^{234}U contributes a very small mass to all uranium mixtures, which means it is not likely to be detectable (this is in contrast to activity, where ^{234}U is most often the largest contributor). However, in all cases, including depleted uranium, ^{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 all claims, with the possible exception of an underestimate.

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 then it is likely that ICP-MS was performed.

Because only two isotopes are reported, the assessment techniques are less complicated than those for alpha spectroscopy results and there is no need for simplifying efficiency methods. However, in order to assess the dose, 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 will account for the majority of the mass except for the case of highly enriched uranium, 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.

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:

$$\frac{6\ \mu g}{0.9927} \times 6.83 \times 10^{-7}\ \mu Ci/\mu g = 4.13 \times 10^{-6}\ \mu Ci \qquad (3\text{-}2)$$

Where 6.83 × 10^{-7} µCi/µg is the specific activity of natural uranium.

Once all results are converted to total uranium activity, they are assessed as usual and can be combined with other sample results that are also in units of total activity.

3.7.3   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) 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 thoron. In these circumstances, the reported thoron values must be converted to dose. DCAS-TIB-0011, *Lung Dose Conversion Factor for Thoron WLM* (NIOSH 2013), contains conversion factors from working level months (WLMs) to dose for respiratory tract segments for ^{220}Rn and ^{219}Rn. Because the exposure model applies specifically to the lung for radon, it also contains conversion factors for ET1 and ET2 for ^{222}Rn.

3.7.4   Assessment of Mixtures of Radionuclides

When dealing with mixtures of materials, such as those discussed in Section 3.2.3.3, all components must be considered and summed for the dose comparison when determining the 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, recycled uranium, 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.

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, recycled uranium contains plutonium and some sites that handled recycled uranium 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.

4.0 ADDITIONAL INTERNAL DOSIMETRY GUIDANCE DOCUMENTS

This section summarizes several documents that provide efficiency methods or additional information for making decisions on exposures to assign to a particular worker. This is intended as an overview only; refer to the individual documents for the current revision and for applicability and limitations of each.

4.1 OCAS-IG-002, INTERNAL DOSE RECONSTRUCTION IMPLEMENTATION GUIDELINE

This document provides implementation guidance on methods and approaches that can be used to reconstruct occupational radiation dose from internally deposited radionuclides in support of EEOICPA (NIOSH 2002).

4.2 OCAS-TIB-009, ESTIMATION OF INGESTION INTAKES

Guidance for estimating associated ingestion intakes when inhalation intakes are assessed through methods other than bioassay (NIOSH 2004).

4.3 DCAS-TIB-011, LUNG DOSE CONVERSION FACTOR FOR THORON WLM

This TIB provides the conversion factor for calculating lung dose from ^{220}Rn decay products measured in working level months (WLM). It also provides conversion factors for ^{222}Rn and ^{219}Rn (NIOSH 2013).

4.4 DCAS-RPT-005, ALTERNATIVE DISSOLUTION MODELS FOR INSOLUBLE PU-238

This report provides guidance on the evaluation of intakes for workers who were exposed to insoluble forms of ^{238}Pu. This guidance applies to Los Alamos National Laboratory, the Mound Site, SRS, and the Nuclear Materials and Equipment Corporation (NUMEC) (NIOSH 2016).

4.5 ORAUT-OTIB-0001, MAXIMUM INTERNAL DOSE ESTIMATES FOR SAVANNAH RIVER SITE (SRS) CLAIMS

This method uses the largest recorded intakes in the history of SRS to assign an overestimated intake for workers (ORAUT 2003a). An average of the largest five intakes for each monitored nuclide was used. Although it is conceivable that an individual had an intake in excess of one of the values, it is very unlikely that the worker had large undocumented intakes of all radionuclides present on the site. Tritium is not included; if the worker had the potential for significant or chronic intakes of tritium, additional dose must be included.

4.6 ORAUT-OTIB-0011, TRITIUM CALCULATED AND MISSED DOSE ESTIMATES

IMBA does not directly calculate intakes from ^{3}H urine data, so an alternative tool was developed. This TIB provides documentation of the method for estimating tritium missed and calculated doses from urine data (ORAUT 2004c).

4.7   ORAUT-OTIB-0014, ASSIGNMENT OF ENVIRONMENTAL INTERNAL DOSES FOR EMPLOYEES NOT EXPOSED TO AIRBORNE RADIONUCLIDES IN THE WORKPLACE

Internal radiation doses to some employees 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 employees, assignment of environmental dose only is appropriate. ORAUT-OTIB-0014 (ORAUT 2004b) provides guidance for determining such instances based on:

- Job description,
- Work location,
- Time frame, and
- Presence or absence of internal monitoring data.

This document also provides information on job categories that can be helpful in determining "significant potential" when assigning coworker intakes.

4.8   ORAUT-OTIB-0018, INTERNAL DOSE OVERESTIMATES FOR FACILITIES WITH AIR SAMPLING PROGRAMS

This method is based on limiting air concentrations (ORAUT 2005c). A chronic exposure to the maximum permissible concentration (MPC) throughout employment is assumed. Because of this, the method applies only to sites that controlled exposure to intakes based on rigorous air sampling programs. While it is possible for a worker to have been occasionally exposed to levels exceeding the MPC, it is very unlikely that an individual was continuously exposed at such levels for 40 hr/wk throughout the employment period. An additional conservatism is achieved by assuming that the airborne activity consisted of the single nuclide, in each year of intake, that results in the largest dose to the organ of interest rather than assigning a mixture of radionuclides.

Application of this overestimating technique is limited to specific sites and organs; refer to the TIB and the tool documentation for applicability.

4.9   ORAUT-OTIB-0022, GUIDANCE ON WOUND MODELING FOR INTERNAL DOSE RECONSTRUCTION

This document focuses on how to use IMBA to evaluate intakes of plutonium by wound, although the concept can be applied to other radionuclides (ORAUT 2005a). Guidance on initial parameters is provided.

4.10   ORAUT-OTIB-0033, APPLICATION OF INTERNAL DOSES BASED ON CLAIMANT-FAVORABLE ASSUMPTIONS FOR PROCESSING AS BEST ESTIMATES

This method (ORAUT 2005d) applies a graded approach to internal dose overestimates and unites the application of ORAUT-OTIB-0014 (ORAUT 2004a), ORAUT-OTIB-0018 (ORAUT 2005c), and coworker dose. Some judgment is needed to apply these values; guidance is provided based on:

- The period of employment,
- The processes at the site,
- The job category and work location, and
- The results of bioassay measurements for the energy employee.

4.11 ORAUT-OTIB-0049, ESTIMATING DOSES FOR PLUTONIUM STRONGLY RETAINED IN THE LUNG

A handful of accidental intakes of plutonium oxides have exhibited long-term retention of plutonium in the lung exceeding that predicted by the standard type S model. This TIB provides assessment methods for calculating a best estimate of the annual organ doses for intakes of type super S plutonium and describes the conditions for applicability of this method (ORAUT 2010).

4.12 ORAUT-OTIB-0054, FISSION AND ACTIVATION PRODUCT ASSIGNMENT FOR INTERNAL DOSE-RELATED GROSS BETA AND GROSS GAMMA ANALYSES

Reactor operations can produce a wide assortment of radionuclides, but routine bioassay monitoring typically includes only those that predominate the mixture or are reported as a gross measurement that could contain an assortment of nuclides. This TIB provides guidance on the assignment of radionuclide-specific intakes of mixed fission and activation products when air sampling or urinalysis data associated with reactors or reactor fuels are available only as gross or total beta activity or gross or total gamma activity (ORAUT 2015). The derived ratios can also be applied to claims where the activity of a single radionuclide in the mixture is known, such as ^{137}Cs from a whole-body count result or ^{90}Sr in a urine sample.

4.13 ORAUT-OTIB-0066, CALCULATION OF DOSE FROM INTAKES OF SPECIAL TRITIUM COMPOUNDS

This document provides guidance on how to use urine bioassay data to calculate best estimates of doses for intakes of organically bound tritium and stable metal tritides.

5.0 ATTRIBUTIONS AND ANNOTATIONS

All information requiring identification was addressed via references integrated into the reference section of this document.

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ATTACHMENT A
RADIATION TYPES BY NUCLIDE FOR ENTRY INTO IREP

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

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

a. E = energy; the radiation effectiveness factors (REFs) for >15-keV electrons and >250-keV photons are equal. To minimize the number of lines in IREP, these two categories can be entered with either energy type and combined if the same distribution type.

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 1994) 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 types
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
43	Technetium	Tc	F, M
44	Ruthenium	Ru	F, M, S
45	Rhodium	Rh	F, M, S

ATTACHMENT B
ICRP PUBLICATION 68 DEFAULT MATERIAL TYPES (continued)

Atomic number	Element	Symbol	Material types
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
96	Curium	Cm	M
97	Berkelium	Bk	M
98	Californium	Cf	M

ATTACHMENT B
ICRP PUBLICATION 68 DEFAULT MATERIAL TYPES (continued)

Atomic number	Element	Symbol	Material types
99	Einsteinium	Es	M
100	Fermium	Fm	M
101	Mendelevium	Md	M

a.   Source: ICRP (1995a).

ATTACHMENT C
RETENTION OF MATERIAL IN THE LUNGS

LIST OF FIGURES

ATTACHMENT C
RETENTION OF MATERIAL IN THE LUNGS (continued)

As noted in Section 3.2.3, material type, also referred to as absorption 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 following an acute inhalation intake is shown in Figure C-1. The material is transferred out of the lungs in less than a tenth of a day. Figures C-2 and C-3 show the clearance of types M and S, respectively, following an acute inhalation intake. The same scale is used for both for comparison.

Figures C-4 through C-6 display the retention and clearance of the three types during and following 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. Types M and S both continue to increase in the lungs over a 365-day chronic intake, with the type M activity reaching about half that of the type S activity. Following 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 type S material in the lungs. Going to a longer intake of 1000 days, it can be seen in Figure C-7 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 C-8, continues to increase throughout the period.

These plots demonstrate that chest counting cannot be used for assessing a type F material because at any time during or following an intake little to no activity is expected to be present in the lungs. Chest counting can be used to detect either type M or S in the lungs but at any given time urine sampling may 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.
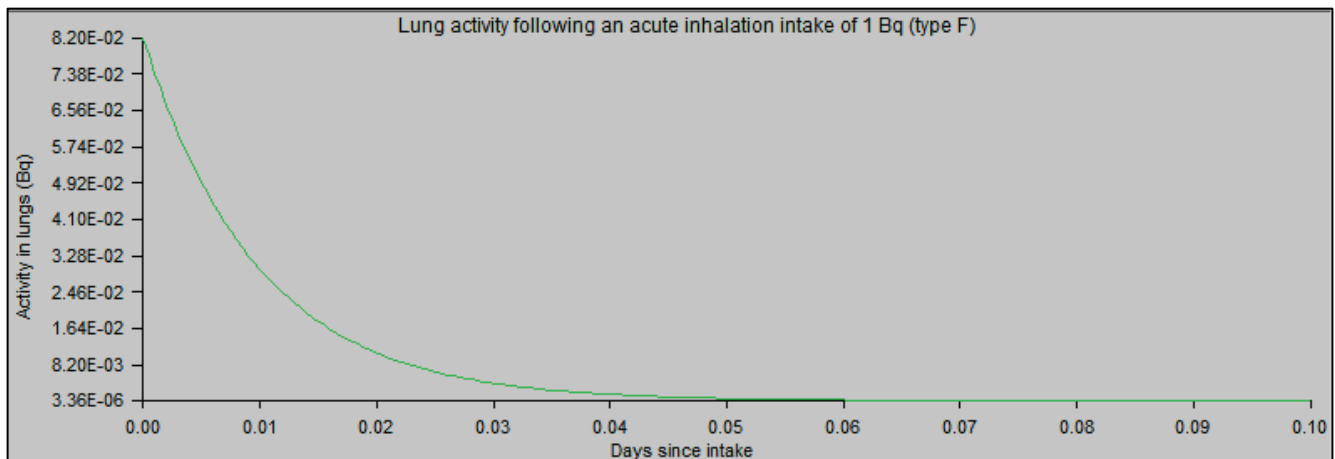

Figure C-1. Lung retention following an acute inhalation intake of 1 Bq of type F material.

ATTACHMENT C
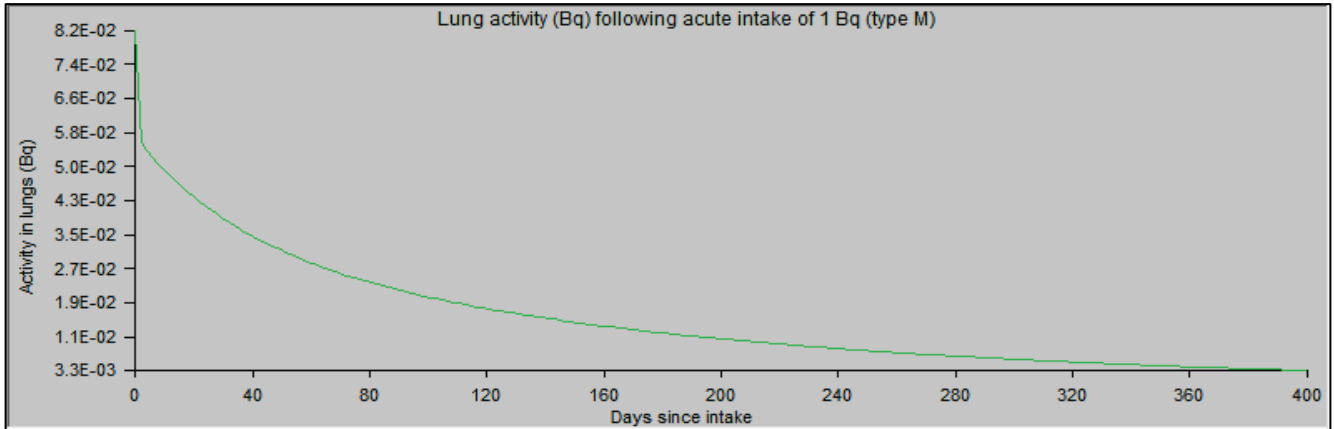**RETENTION OF MATERIAL IN THE LUNGS (continued)**

Figure C-2. Lung retention following an acute inhalation intake of 1 Bq of type M material.
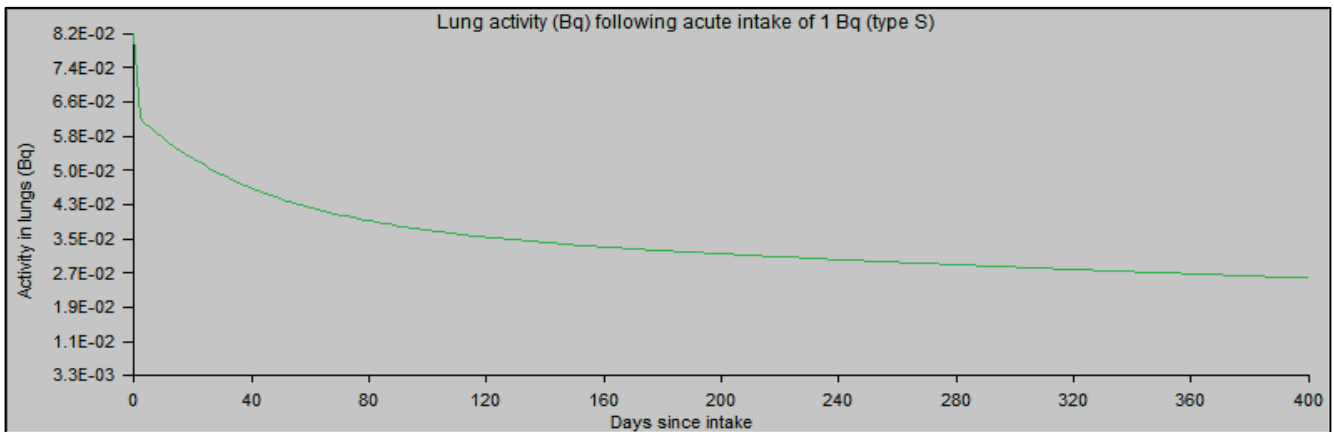

Figure C-3. Lung retention following an acute inhalation intake of 1 Bq of type S material.
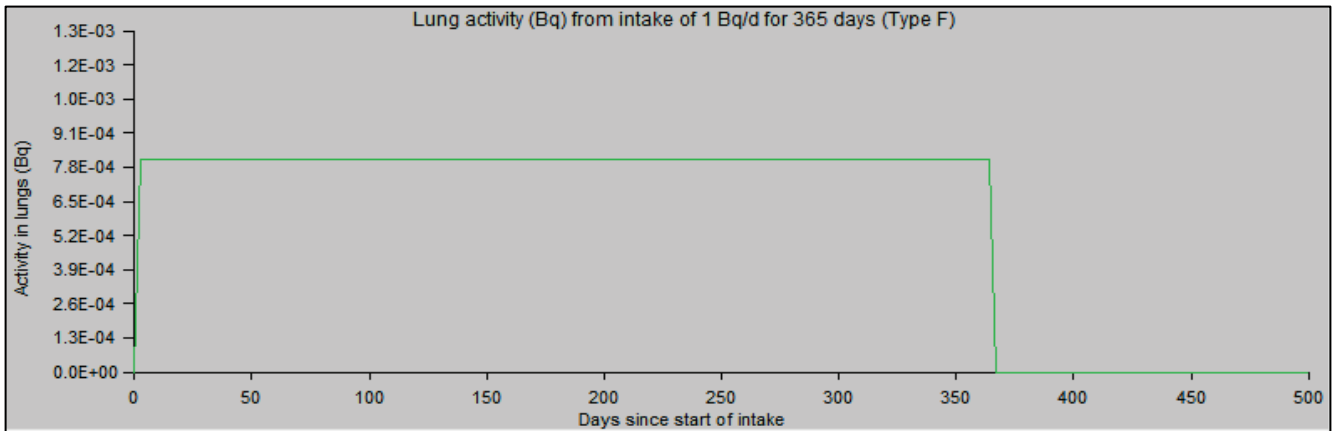

Figure C-4. Lung retention of a 365-d chronic inhalation intake of a type F material.

ATTACHMENT C
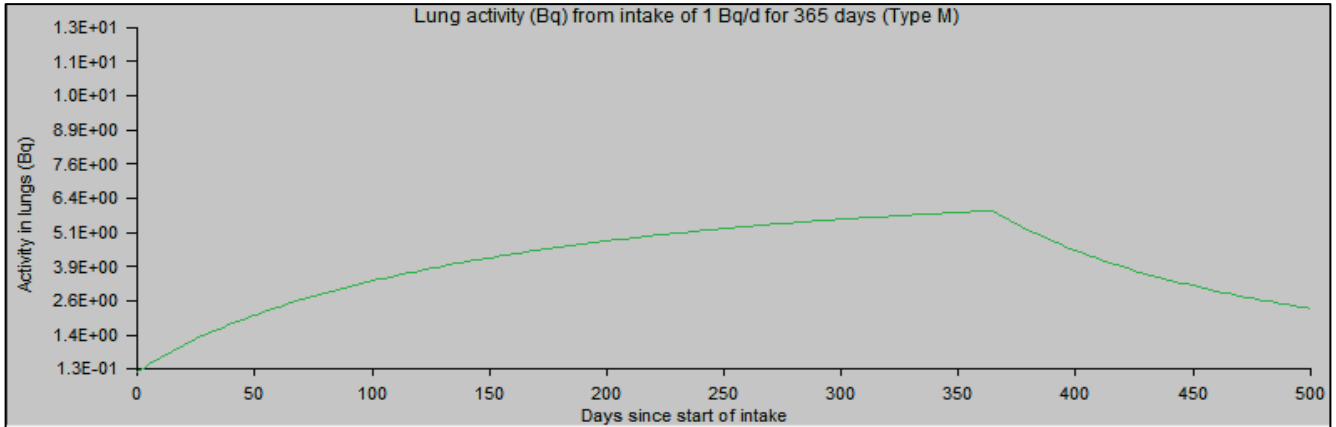**RETENTION OF MATERIAL IN THE LUNGS (continued)**

Figure C-5. Lung retention of a 365-d chronic inhalation intake of a type M material.
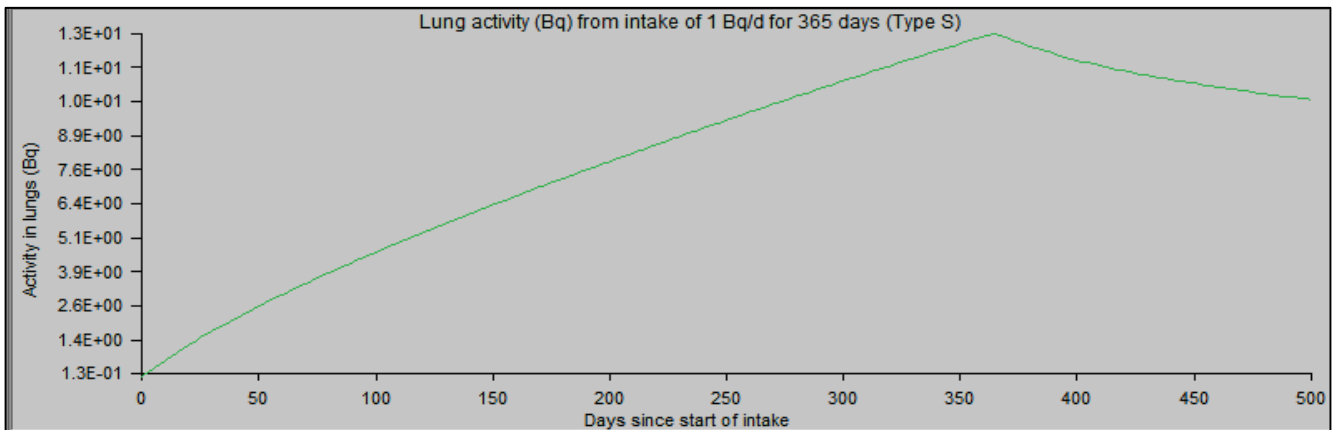

Figure C-6. Lung retention of a 365-d chronic inhalation intake of a type S material.
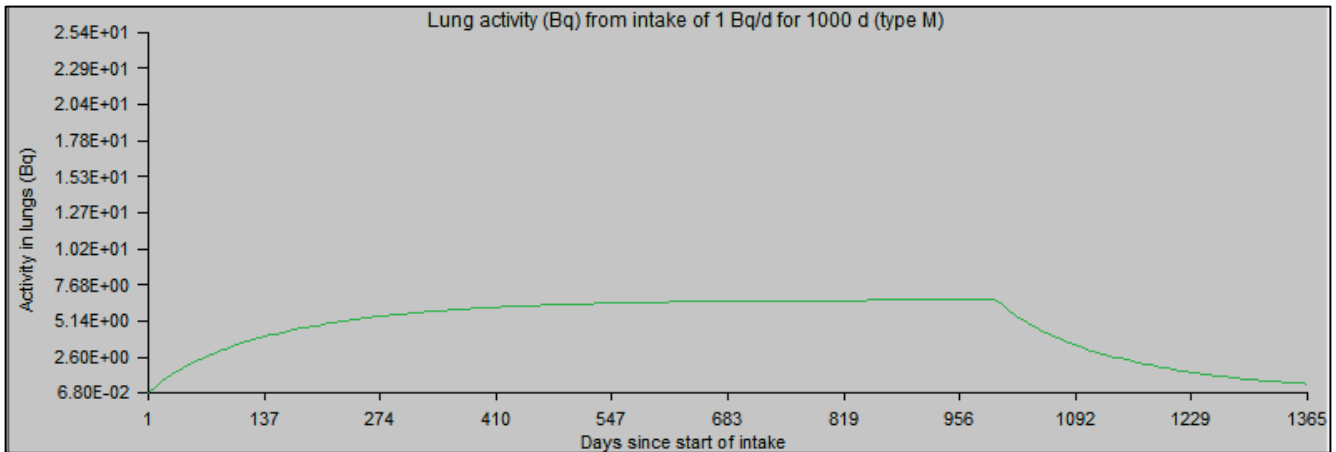

Figure C-7. Lung retention of a 1,000-d chronic inhalation intake of a type M material.

ATTACHMENT C
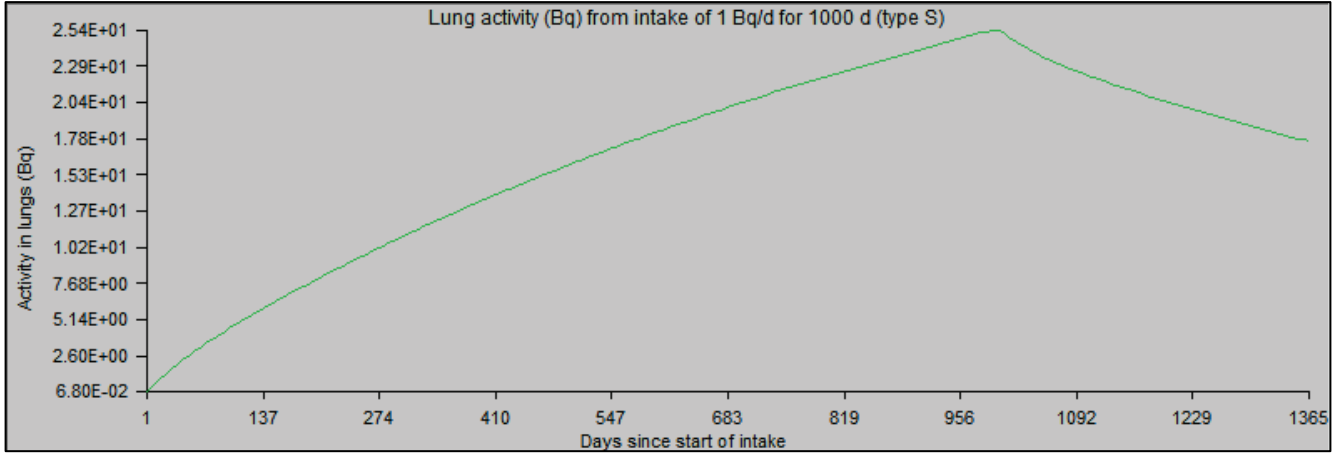**RETENTION OF MATERIAL IN THE LUNGS (continued)**

Figure C-8. Lung retention of a 1,000-d chronic inhalation intake of a type S material.