SAND2014-17441R

Sandia National Laboratories	Procedure No: DTBD006 Issue No: 04 Effective Date: 9/9/2013

RADIATION PROTECTION DOSIMETRY PROGRAM TECHNICAL BASIS DOCUMENT

SANDIA NATIONAL LABORATORIES INTERNAL DOSIMETRY TECHNICAL BASIS MANUAL

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ABSTRACT

The Sandia National Laboratories' Internal Dosimetry Technical Basis Manual is intended to provide extended technical discussion and justification of the internal dosimetry program at SNL. It serves to record the approach to evaluating internal doses from radiobioassay data, and where appropriate, from workplace monitoring data per the Department of Energy Internal Dosimetry Program Guide DOE G 441.1C. The discussion contained herein is directed primarily to current and future SNL internal dosimetrists. In an effort to conserve space in the TBM and avoid duplication, it contains numerous references providing an entry point into the internal dosimetry literature relevant to this program. The TBM is not intended to act as a policy or procedure statement, but will supplement the information normally found in procedures or policy documents.

The internal dosimetry program outlined in this manual is intended to meet the requirements of Federal Rule 10CFR835 for monitoring the workplace and for assessing internal radiation doses to workers. Many of the recommendations from the following DOE documents are incorporated into this program:

- Radiation Protection Program Guide
- DOE Internal Dosimetry Standard
- DOE Plutonium Good Practices Manual
- DOE Uranium Good Practices Manual

ACKNOWLEDGMENTS

The Internal Dosimetry program is eternally grateful to Charles A. Potter, who authored the previous revision, as well as Thomas R. LaBone, who authored the original Savannah River Site Internal Dosimetry Technical Basis Manual.

The following authors of the 13th revision of the Savannah River Site Internal Dosimetry Technical Basis Document on which this revision is partially based are also acknowledged:

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PREFACE

The Internal Dosimetry Technical Basis Manual (TBM), by default, is a retrospective on the scientific foundation of the program for which it was written. The TBM is essentially a "snapshot" of the technical basis as of the day the manual is signed for approval. It does not, and should not, enter into discussions of anticipated changes but should only reflect current policies. Any improvements and/or changes in the Internal Dosimetry program made subsequent to the current revision are captured in Radiation Protection Dosimetry Program numbered documents which then form the basis for program and/or procedure changes. The changes captured by the numbered documents will be incorporated in the next revision of the TBM. As such it is not unexpected for the TBM to lag program improvements and procedural guidance.

NOMENCLATURE

AEC	Atomic Energy Commission
ACL	administrative control level
AEDE	annual effective dose equivalent
ALI	annual limit on intake
AMAD	activity median aerodynamic diameter
AMS	Alpha Management System
AMTD	activity median thermodynamic diameter
ANSI	American National Standards Institute
APF	assigned protection factor
ARA	airborne radioactivity area
BEIR	Biological Effects of Ionizing Radiation
Bq	Becquerel
CA	contamination area
CAM	continuous air monitor
cc	cubic centimeter
pCi	picoCurie (E-12 Ci)
nCi	nanoCurie (E-09 Ci)
µCi	microCurie (E-06 Ci)
Ci	Curie
Ci/g	curie(s) per gram
CED	committed effective dose
CEDE	committed effective dose equivalent (predecessor to CED)
CEqD	committed equivalent dose
CFR	Code of Federal Regulations
CSU	combined standard uncertainty
d	day
DAC	derived air concentration
DARB	Dose Assessment Review Board
DCF	dose conversion factor
DIL	derived investigation level
DL	decision level
DOE	Department of Energy
DOELAP	Department of Energy Laboratory Accreditation Program
dpm	disintegration per minute
DPSOP	DuPont Standard Operating Procedure
DTPA	diethylene triamine pentaacetic acid
DU	depleted uranium
E(50)	50-year committed effective dose
eV	electron volt

keV	kilo electron volt
MeV	mega electron volts
EDTA	ethylene diamine tetraacetic acid
EPA	United States Environmental Protection Agency
ERDA	Energy Research and Development Administration
ET	extrathoracic
f1	fractional absorption in the gastrointestinal tract
µg	microgram
mg	milligram
g	gram
kg	kilogram
G	Gray
GAS	general air sampler
GI	gastrointestinal
Ht(50)	50-year committed organ or tissue equivalent dose
hr	hour
HT	elemental tritium
HEU	highly enriched uranium
HPGe	high purity germanium
HPS	Health Physics Society
HRTM	Human Respiratory Tract Model
HTO	tritiated water vapor or liquid
IAEA	International Atomic Energy Agency
ICRP	International Commission on Radiological Protection
IH & RP	Industrial Hygiene and Radiation Protection Organization
IMBA	Integrated Modules for Bioassay Analysis
IRF	intake retention fraction (or function)
IVCF	In Vivo Counting Facility, Building 735-4B
L	liter
LANL	Los Alamos National Laboratory
lb	pound
Lc	critical level
LLD	lower limit of detection
LSC	liquid scintillation counter
µm	micrometer; also known as micron
m	meter
mL	milliliter
MDA	minimum detectable activity
MDC	minimum detectable concentration
MDD	minimum detectable dose

MDI	minimum detectable intake
MOW	Member(s) of the Workforce
MPBB	maximum permissible body burden
MPC	maximum permissible concentrations
MPD	most probable date (of intake)
MPLB	maximum permissible lung burden
mrem	millirem
NaI	sodium iodide
NALI	non-stochastic (or deterministic) annual limit on intake
NBS	National Bureau of Standards
NCRP	National Council on Radiation Protection and Measurements
NIOSH	National Institute for Occupational Safety and Health
NRC	Nuclear Regulatory Commission
OBT	organically bound tritium
ORNL	Oak Ridge National Laboratory
OSHA	Occupational Safety and Health Administration
PAAA	Price Anderson Amendment Act
PAPR	powered air purifying respirator
PAS	personal air sampler
ppm	parts per million
PUREX	Plutonium-Uranium Extraction process
QA	quality assurance
QC	quality control
RBM	red bone marrow
REM	roentgen equivalent man
ROI	region of interest
RP	Radiation Protection
RPP	Radiation Protection Program
RPDP	Radiation Protection Dosimetry Program
RPPM	Radiological Protection Procedures Manual (MN471016)
RU	recycled uranium
RWF	radiation weighting factor
RWP	radiological work permit
s	second
SALI	stochastic annual limit on intake
SEE	specific effective energy
SMT	stable metal tritide
SNL	Sandia National Laboratories
SRS	Savannah River Site

STC	special tritium compounds
Sv	Sievert
TBM	technical basis manual
TLV	threshold limit value
TPU	total propagated uncertainty
TWF	tissue weighting factor
ULI	upper large intestine
USLF	unweighted least squares fit
UNSCEAR	United Nations Scientific Committee on the Effects of Atomic Radiation
USTUR	United States Transuranium and Uranium Registries
WHO	World Health Organization
WLSF	weighted least squares fit
y	year
dB	decibel
DOE	Department of Energy
SNL	Sandia National Laboratories

1. INTRODUCTION

1.1. Program Purpose

The SNL Internal Dosimetry Program is intended to meet the requirements of the Federal Rule 10CFR835 for monitoring the workplace and assessing internal radiation doses to workers[1]. In order to meet the intent of those requirements, many recommendations from the following DOE documents are incorporated into our program:

- Radiation Protection Program Guide[2]
- DOE Internal Dosimetry Standard[3]
- DOE Plutonium Good Practices Manual[4]
- DOE Uranium Good Practices Manual[5]

The objectives of the internal dosimetry program are to detect, assess, and document occupational intakes of radioactive material. This complex and comprehensive program validates and verifies the integrity of both the physical confinement features designed into SNL facilities and the administrative controls documented in Radiation Protection policies and procedures, as well as technical work documents involving radiological work.

1.2. Program Overview

The internal dosimetry program at SNL is administered by the Radiation Protection Dosimetry Program within the Internal Hygiene and Radiation Protection Laboratories department. The Radiation Protection department and the IH & RP Labs department together constitute the Sandia Radiation Protection Program, or RPP[i] [ii]. The internal dosimetry program achieves its objectives through the use of the following radiation protection elements:

- Workplace and personal contamination monitoring
- Air monitoring
- Radiobioassay
- Dose evaluation

[i] As required in 10CFR835.101
[ii] The Sandia RPP and associated requirements are described in the *Radiological Protection Procedures Manual*, MN471016

- Administrative scheduling and reporting
- Regulation and oversight

Both contamination and air monitoring provide data used to determine if workers have been exposed to radioactive materials. Contamination monitoring is provided by the SNL Radiation Protection program while air monitoring is prescribed through the use of a procedure collaboratively produced by Dosimetry and RP[6]. Radiobioassay is prescribed by RPDP dosimetrists and analyzed by onsite or offsite service laboratories in accordance with a RPDP controlled Statement of Work. Much like contamination and air monitoring, radiobioassay provides data that is used by RPDP to determine if workers have been exposed to radioactive materials. However, this radiobioassay data is also used by the dose evaluation element to evaluate intakes of radioactive materials. This evaluation element also serves to assess the ability of the radiobioassay techniques to detect intakes and to design appropriate radiobioassay schedules. Regulatory and oversight elements ensure that the program continues to meet objectives through performance testing and assessments.

1.3. New in Revision 4 of the TBM

- It should be noted that the entire TBM (previously titled the *Technical Basis Document for Internal Dosimetry at Sandia National Laboratories*) has been completely rewritten using both the previous revision and the Savannah River Site Internal Dosimetry Technical Basis Manual. [7]
- Addition of a disclaimer pertaining to the purpose of a technical basis manual
- Addition of a list of relevant Acronyms and Abbreviations
- Addition of a list of relevant Definitions

2. SOURCE TERM

Activities at SNL have the potential to expose MOW to a variety of radionuclide compounds. However, the majority of internal exposures at SNL are expected to result from a select subset of radionuclides. The proper selection and interpretation of bioassay techniques requires an understanding of the physical, radiological, and biologic properties of these radionuclides.

For the purposes of this revision of the TBM, the primary radionuclides used at SNL will be considered to be tritium, the isotopes of uranium, the isotopes of plutonium, strontium, and americium based on the previous draft and historical experience.

Occupational exposures may less frequently result from exposures to additional radionuclides not specifically addressed in this technical basis document. The bioassay philosophies used to monitor tritium, uranium, and plutonium exposures are anticipated to be adequate in monitoring exposures to other potential contaminants at SNL. Exceptions will be addressed, as needed, in site-specific technical basis document(s) in the dosimetry program.

2.1. Radon

Radon is not considered to be within the scope of this manual. All radon exposures are considered to be non-occupational and therefore no radiobioassay is performed. This position will be reassessed if future operations at SNL involve exposure to radon other than from naturally occurring sources.

2.2. Tritium

Tritium (^{3}H or T) is the only radioactive species of the three major isotopes of hydrogen. It decays by beta emission with an average energy of 5.7 KeV over a radiological half-life of 12.3 years. Tritium may be encountered in numerous physical and chemical forms since its chemical properties are almost identical to H which is ubiquitous in the environment. Tritium is normally encountered at SNL as tritium gas, in the form of HT, DT, or T_2, and tritiated water, in the form of HTO, DTO, or T_2O.

2.2.1. Facilities Using Tritium

The following facilities process, store, or are potentially contaminated with tritium:

TA I: Building 891 - Facilities containing erbium tritide (ErT_3)

TA II: Tritium may be contained within test components which can be converted to DT or HTO.

TA III: Tritium is a potential contaminant at some environmental restoration sites (e.g., Mixed Waste Landfill, etc.)

Radioactive and Mixed Waste Management Facility (RMWMF) – Any radionuclides used at SNL including ^{3}H, U, Th, Pu, Am, Sr and tracer quantities of others.

TA IV: PBFA-II and SABER facilities have used accelerator targets containing tritium.

TA V: ACRR, Hot Cell Facility, SPR facilities are potentially contaminated by tritium compounds.

CTF: (none known)

TTR: (none known)

2.3. Uranium

Occupational exposures to isotopes of uranium at SNL mainly occur from natural uranium (U-Nat), depleted uranium (DU), and enriched uranium (EU) containing different proportions of the naturally occurring uranium isotopes (i.e., ^{234}U, ^{235}U, and ^{238}U).

Table 1, Significant Uranium Decay Series Radionuclides at SNL

Uranium Decay	Radionuclide	Energies (MeV) and % Abundances of Major Emissions		
Series Member	Half-Life	Alpha	Beta	Gamma
^{238}U	4.51x10^{9} y	4.15 (25%) 4.20 (75%)	(none)	(none)
^{234}Th	24.1 d	(none)	0.103 (21%) 0.193 (79%)	0.063 (3.5%) 0.093 (4%)
^{234m}Pa	1.17m	(none)	2.29 (98%)	0.765 (0.3%) 1.001 (0.6%)

Progeny of these isotopes are radioactive and form decay chains. Uranium-238 and ^{234}U are members of the uranium decay series, while ^{235}U is a member of the actinium decay series. Several of these progeny may have significant internal dosimetry implications when secular equilibrium is maintained. However, most of the uranium forms historically encountered at SNL have been chemically extracted from the virgin feed materials.

Progeny with long half-lives (e.g., ^{234}U in the uranium series, and ^{231}Pa in the actinium series) effectively prevent secular equilibrium along the entire decay chain in these cases. The resultant radionuclide progeny which occur in significant abundance to impact radiological controls are ^{234}Th and ^{234m}Pa in the uranium series, and ^{231}Th in the actinium series (EGG 88). However,

other decay progeny may be present from incomplete chemical separations and from naturally occurring deposits of uranium. Table 1 and Table 2 present the radiological characteristics of common uranium isotopes and major progeny at SNL.

Table 2, Significant Actinium Decay Series Radionuclides at SNL

Actinium Decay	Radio-nuclide	Energies (MeV) and % Abundances of Major Emissions		
Series Member	Half-Life	Alpha	Beta	Gamma
^{235}U	$7.04x10^{8}$ y	4.37 (18%)	(none)	0.144 (11%)
		4.40 (57%)		0.185 (54%)
		4.58 (8%)		0.204 (5%)
^{231}U	25.5 h	(none)	0.14 (45%)	0.026 (2%)
			0.22 (15%)	0.084 (10%)
			0.305 (40%)	

2.3.1. Facilities Using Uranium Compounds

The following facilities process, store, or are potentially contaminated with uranium compounds:

TA I: (none known)

TA II: (none known)

TA III: Uranium compounds are potential contaminants at some environmental restoration sites. Uranium may also be found in materials processing at the Radioactive and Mixed Waste Management Facility (RWMWF).

Manzano Storage Bunkers: Waste storage containing uranium.

TA IV: Uranium targets may be used in Z-Machine.

TA V: ACRR, Hot Cell Facilities, and SPR facilities process or are contaminated by uranium compounds.

CTF: Uranium compounds have been used for testing purposes, and uranium compounds are potential contaminants at some environmental restoration sites throughout the area.

TTR: Uranium compounds are potential contaminants at some environmental restoration sites throughout the area.

2.4. Plutonium

Plutonium exposures at SNL are expected to be rare compared to other exposure sources. The principle isotopes of plutonium found in non-production DOE facilities like SNL are ^{238}Pu and

^{239}Pu. Compounds of these isotopes potentially contain other radionuclides; however these contaminants are typically found in minute quantities and are subsequently ignored in bioassay programs not connected to an intake. The exception is ^{241}Am, which is discussed in Section 2.6.

2.4.1. Facilities Using Plutonium

The following facilities process, store, or are potentially contaminated with plutonium compounds:

TA I: (none known)

TA II: (none known)

TA III: Plutonium may be found in the Radioactive and Mixed Waste Management Facility (RWMWF) materials processing.

TA IV: Plutonium targets may be used in Z-Machine.

TA V: ACRR and Hot Cell Facilities may be contaminated by plutonium compounds.

CTF: (none known)

TTR: Plutonium compounds may be present at some environmental restoration sites.

2.5. Strontium

While not widely used, the principle isotope of strontium found at SNL is ^{90}Sr.

2.5.1. Facilities Using Strontium

The following facilities process, store, or are potentially contaminated with strontium:

TA I: (none known)

TA II: (none known)

TA III: Strontium may be found in the Radioactive and Mixed Waste Management Facility (RWMWF) materials processing.

TA IV: (none known)

TA V: ACRR and Hot Cell Facilities may be contaminated by strontium compounds.

CTF: (none known)

TTR: Strontium compounds may be present at some environmental restoration sites.

2.6. Americium

Compounds of Plutonium isotopes ^{238}Pu and ^{239}Pu potentially contain ^{241}Am, which is of significant note for in-vivo bioassay application.

The 59.5 KeV gamma ray emission from ^{241}Am is easy to detect and can be used to quantify smaller exposures of plutonium when the ^{241}Am/Pu ratio is known.

2.6.1. Facilities Using Americium

The following facilities process, store, or are potentially contaminated with americium compounds:

TA I: (none known)

TA II: (none known)

TA III: Americium may be found in materials processing at the Radioactive and Mixed Waste Management Facility (RWMWF).

TA IV: Plutonium targets possibly used in Z-Machine may contain ^{241}Am

TA V: (none known)

CTF: (none known)

TTR: (none known)

3. MONITORING METHODS

An internal dosimetry monitoring program is considered to consist of one or more of the following monitoring methods:

- Workplace air monitoring, which identifies releases of radioactive material into the air of the workplace,
- Breathing zone air monitoring, which provides an estimate of the exposure to the worker, and
- Radiobioassay, which measures the quantity of radioactive material in the body or excreta of the worker.

This manual is concerned primarily with radiobioassay for Sandia MOW assigned to perform radiological work with reasonable or likely potential for intake of specified radionuclides.

3.1. Air Monitoring

Sandia's air monitoring program is discussed in detail in the *Technical Basis and Expectations for Implementing Air Sampling/Monitoring at Sandia National Laboratories*. [8] This program is based on NUREG-1400[9] methodology designed to meet the requirements of the federal rules 10CFR835.403 and 10CFR835.601, as well as meet the objective of air monitoring[iii] by measuring the concentrations of airborne radioactive material to assess worker exposure to:

- Assess worker exposure to airborne radioactive material
- Determine posting requirements
- Determine the need for and prescribe appropriate personnel protection (respiratory protection in particular) from airborne radioactive material
- Determine the effectiveness of the protective equipment and measures after the fact
- Provide warnings of significantly elevated levels of airborne radioactive materials and
- Determine the effectiveness of the confinement of radioactive material.

[iii] Section 5.3, *Radiation Protection Programs Guide for Use with Title 10, Code of Federal Regulations, Part 835, Occupational Radiation Protection*, DOE G 441.1-1C, May 2008

The objective of air monitoring is met through the use of a variety of air sampling devices, selected depending on the need for each of these measurements to be *representative* of the concentration of aerosols in the air breathed by a worker for the entire duration of the exposure and in accordance with the procedure. [10]

Typically personal air sampler (PAS) results are used to estimate intakes to a worker because they are representative by definition, but other types of air samplers can be used. Samplers that are less representative than a PAS are used to estimate intakes only in unusual situations where no superior information on the exposure is available.[9]

Air samplers such as retrospective[iv] stationary air samplers and continuous air monitors (CAMs) are unlikely to be representative for a specific worker and should therefore not be used to assign an intake whenever possible.

Note that if a PAS is used, likelihood of an intake of radioactive materials will be evaluated and assigned to the person as necessary in accordance with the discussion in Chapter 0.

3.1.1. Personal Air Samplers

PAS has the following characteristics that make it very attractive for bioassay monitoring:

- When viewed as a radiobioassay technique, PAS has a very high intake-retention fraction (IRF) that results in a low MDD. For example, a 24-hour urine sample started immediately after an intake of Type S ^{239}Pu will contain 2.35×10^{-6} of the total quantity of material inhaled. In other words, the intake retention fraction (IRF) is 2.35×10^{-6}. The PAS filter will contain a fraction[v] of the total quantity of material inhaled equivalent to 0.2, which gives it an IRF of 0.2. This approach, discussed in detail by Skrable[11] et al. , results in an MDD for PAS that is at least 500 times lower than that for urine radiobioassay under the stated conditions.

- The advantage of PAS, over urine radiobioassay, increases as the solubility of the inhaled material decreases.

- PAS filters are initially analyzed by gross alpha and gross beta counting methods, whereas *in vitro* samples are analyzed by element or nuclide-specific analytical methods. This means the source term must be known when urine radiobioassay is prescribed[vi] but not for PAS.

- PAS filters can be screened within minutes and problems identified in a timely fashion, while special urine samples may take 5-10 days to analyze.

[iv] *Retrospective* means that the collection filter is not analyzed in real-time like a CAM.
[v] 4 lpm / 20 lpm = 0.2 (difference in "breathing rate" of PAS vs. reference man)
[vi] The alternative of analyzing the urine sample for "everything" increases the chances of running into false positive results, which can be rather problematic to deal with.

- PAS is very useful for monitoring occupational intakes of materials like uranium and thorium which have the potential for significant natural interference in *in vitro* samples.

The primary drawback of PAS is that, to detect an intake, it must be worn and running when the exposure occurs, i.e., it must be prescribed before the fact (prospectively). Radiobioassay can always be prescribed after the exposure (retrospectively). Other, less important disadvantages of PAS are that it may not correlate well with radiobioassay[12,13] and that the worker has the added burden of wearing the equipment.

Situations where PAS should be considered include:

- Monitoring workers for the purpose of assigning dose when radiobioassay alone is not adequate. Examples of this may be monitoring workers:
    - For exposure to SMTs where urine radiobioassay alone cannot be used to distinguish intakes of tritides from tritium oxides
    - With potential exposure to thorium, where the natural thorium background in urine makes it difficult to identify occupational exposures with radiobioassay alone
    - For potential exposure to actinides and/or insoluble compounds whose MDD is lower for standard urinalysis than for PAS
- Performing validation sampling to back up a decision not to assign respiratory protection for tasks that historically had respiratory protection assigned. Examples of this include:
    - When analysis shows that significant airborne radioactivity is unlikely and the decision is made not to assign respiratory protection[14,15]. PAS may be used for a specified time period to validate this decision[16].
- Monitoring the effectiveness of supplied air respiratory protection used in environments where it is not feasible to place a CAM. The best example of this would be:
    - The placement of a PAS inside anti-contamination clothing with the worker when working in a highly contaminated area.

3.2. Radiobioassay

The radiobioassay methods used at SNL to detect and quantify radioactive material in the human body and excreta are summarized in this section. Radioactive decay data and minimum detectable amounts (MDAs) for various radiobioassay methods are given in this section. These MDAs are used in Chapter 4 to estimate the ability of a radiobioassay technique to identify intakes. Unless otherwise noted, the radioactive decay information presented in this chapter for fission/activation products is taken from Kocher[17] and the information for actinides is taken from ICRP 38[18].

3.2.1. Specification of Minimum Detectable Amount[vii]

The MDA quoted for each radiobioassay method is that quantity of material that has a 5% chance of not being detected (i.e., getting a false negative) under the assay conditions indicated[viii]. The MDA (minimum detectable concentration for urine, or MDC) is used to design and measure the ability of internal dosimetry programs. The MDA is not used to decide if a specific analysis has or has not detected activity above background. The decision level is used for this purpose (see Chapter 0).

3.2.2. Radiobioassay Services

A summary of the types of *in vitro and in vivo* samples typically collected in the SNL radiobioassay program are listed in the following table.

[vii] The MDAs given for urine radiobioassay are technically minimum detectable concentrations (MDCs), but for these discussions they will be referred to as MDAs.
[viii] The MDAs given in this chapter for an *in vitro* analytical method are two times the average decision level for that analytical method. The MDAs for *in vivo* methods are those reported by the ABACOS 2K software.

Table 3, Bioassay Sample Types

Designation	Sample Volume	Sample Volume Processed	TAT (days)	Bioassay Sample Code	Lab
Baseline or Termination (Verification) Urine, alpha or beta spec	2000 mL	1000 mL	30	I	Offsite
Urine, alpha or beta spec	2000 mL	1000 mL	30	I	Offsite
24-hour Urine, alpha or beta spec	Time dependent volume	1/2 of sample volume	30	I	Offsite
Positive Urine Follow-Up, alpha or beta spec	Time dependent volume	Entire volume	30	I	Offsite
nconclusive Analysis Urine Follow-Up, alpha or beta spec	2000 mL sample	1000 mL	30	I	Offsite
TIMS Urine	1000 mL sample	500 mL	30	I	LANL
Baseline, or Termination Tritium Urine	>50 mL	5 mL aliquot	14	S	RPSD
Urgent Tritium Urine	>50 mL	5 mL aliquot	1	S	RPSD
Fecal	single void > 50 g	entire dissolved sample	30	F	Offsite
Whole Body Count	N/A	N/A	30	W	RPSD
Wound Count	N/A	N/A	1	N/A	RPSD

TAT refers to the turn-around time, in calendar days, for completing an analysis and providing all data deliverables to RPDP. Once fecal samples are digested, the same radiochemical separation and counting techniques that are used for urine samples are employed.

3.2.3. In Vivo

The onsite Radiation Protection Sample Diagnostics (RPSD) Program laboratory has traditionally been used for all *in vivo* radiobioassay for the duration of the internal dosimetry program. Whole body counts are performed using a Canberra Accuscan™ with two Ge detectors for whole body counts. A count is 10 minutes in duration. ^{137}Cs is used as the marker radionuclide for intake evaluation. Follow-up counts are performed if ^{137}Cs activity is above the DL criteria discussed in the intake evaluation Section 0, or if the signature gamma-emitters of any of the radionuclides in the RPDP SOW are detected. The MDA for ^{137}Cs is 12,000 pCi/L.

RPSD maintains a wound counter for wound bioassay analysis. A Canberra Broad Energy Germanium (BEGe) n-type detector and analysis protocols are also maintained by RPSD for wound counting.

3.2.4. In Vitro

In vitro radiobioassay for alpha spectroscopy and beta counting have been performed by an offsite service provider throughout the history of the SNL internal dosimetry program. RPDP has used its currently-contracted radiobioassay services provider since 2002. For analysis of alpha-emitting actinides, the analyte of interest is isolated from an aliquot filtered by ion exchange and mounted by electro-deposition or co-precipitated onto a filter, and analyzed on a silicon surface-barrier alpha spectrometer. The typical count time is 1000 minutes. Samples that contain activity above the decision level are reported to RPDP and are recounted for a count time of 2500 minutes. ^{90}Sr analyses are performed by ion exchange followed by a 1000-minute count on a gas-flow proportional counter. Samples that contain activity above the decision level are reported to RPDP and are recounted for a count time of 1000 minutes.

RPSD performs all *in vitro* tritium analysis by direct aliquot and liquid scintillation counting for 30 minutes. ICP-MS was replaced by alpha spec as the RPDP default uranium analysis due to a technical shortfall of ICP-MS in measuring uranium-235, and the increased need to analyze for this constituent in various SNL waste streams.

3.2.5. Radiobioassay Methods for Selected Radionuclides

Radiobioassay methods for the following elements, listed in order of descending atomic number, are given in this section:

- americium
- plutonium
- uranium
- cesium
- strontium
- cobalt
- tritium

3.2.5.1. Method MDAs

If method MDAs are not listed in each of the following sections, they are provided in Table 7.

3.2.5.2. Americium-241

^{241}Am decays with a half-life of 432 years to ^{237}Np by alpha particle emission. Because of the difference in half-lives between ^{241}Am and ^{237}Np, there is no significant in-growth of radioactive daughters. The energies and intensities of the two primary alpha particles emitted are

5443 keV	12.8%
5486 keV	85.2%

^{241}Am emits the following low-energy photons, which may be useful for radiobioassay:

11.9 keV	0.808%
13.9 keV	13.2%
15.9 keV	0.505%
17.7 keV	19.5%
20.9 keV	4.8%
26.4 keV	2.4%
59.5 keV	35.9%

Urine radiobioassay via standard alpha spectrometry is the typical method for analysis. The method MDA is provided in Table 7. Wound counting has an MDA of 3.31E-4 µCi.

3.2.5.3. Plutonium-241

^{241}Pu decays with a half-life of 14.4 years to ^{241}Am by beta particle emission. The beta particles emitted have a maximum energy of 21 keV. No photons useful for radiobioassay are emitted by ^{241}Pu. Currently, no radiobioassay sampling is performed for ^{241}Pu. The analysis is retained for possible future waste processing projects. The method MDA is provided in Table 7.

3.2.5.4. Plutonium-240

^{240}Pu has a half-life of 6537 years and decays to ^{236}U by alpha particle emission. Because of the difference in half-lives between ^{240}Pu and ^{236}U, there is no significant in-growth of radioactive daughters. The energies and intensities of the two primary alpha particles emitted are

5124 keV	26.5%
5168 keV	73.4%

Plutonium-240 emits the following low-energy photons that may be useful for radiobioassay

11.6 keV	0.186%
13.6 keV	3.77%
15.4 keV	0.109%
17.1 keV	5.32%
20.3 keV	1.22%
45.2 keV	0.045%

Urine samples are analyzed by alpha spectrometry, which cannot differentiate between ^{239}Pu and ^{240}Pu. Therefore, the results of ^{240}Pu urine radiobioassay are reported as ^{239}Pu/^{240}Pu. Urine radiobioassay via standard alpha spectrometry is the typical method for analysis. The method MDA is provided in Table 7.

3.2.5.5. Plutonium-239

^{239}Pu has a half-life of 24065 years and decays to ^{235}U by alpha particle emission. Because of the difference in half-lives between ^{239}Pu and ^{235}U, there is no significant in-growth of radioactive daughters. The energies and intensities of the three primary alpha particles emitted are:

5105 keV	10.7%
5143 keV	15.2%
5156 keV	73.8%

^{239}Pu emits the following low-energy photons that may be useful for radiobioassay:

11.6 keV	0.073%
13.6 keV	1.48%
15.4 keV	0.042%
17.1 keV	2.09%
20.3 keV	0.486%
51.6 keV	0.0208%

SNL's standard ^{239}Pu analysis protocol is urine alpha spectrometry, which cannot differentiate between ^{239}Pu and ^{240}Pu. The results of urine radiobioassay are reported as ^{239}Pu/^{240}Pu, which includes any ^{240}Pu that may be present. ^{239}Pu is seldom found without other plutonium isotopes such as ^{238}Pu, ^{240}Pu, and ^{241}Pu. 6% weapons grade and 12% fuels grade plutonium have been used in the past at SNL. The two mixtures are described below by the percent of ^{240}Pu they contain by mass: 6% Pu (6% ^{240}Pu), and 12% Pu (12% ^{240}Pu)[19] (see Table 5-1).

Table 4, Activity fractions for 6% and 12% plutonium

	6 % Pu Activity Fraction	**12% Pu Activity Fraction**
^{241}Pu	0.8367	0.9712
^{240}Pu	0.0278	0.0045
^{239}Pu	0.1181	0.0030
^{238}Pu	0.0174	0.0212

For work that deals exclusively with ^{239}Pu, SNL has used Thermal Ionization Mass Spectrometry (TIMS) for ^{239}Pu analysis via a radiobioassay service contract with Los Alamos National Laboratories (LANL). This highly sensitive isotope mass spectrometry technique has the capability of detecting extremely low levels of ^{239}Pu in urine. Due to its high cost compared with the standard alpha spectrometry method and lead time required for LANL sample kit procurement, TIMS is performed on an as-requested basis and is not included in the standard analytical methods funded by RPDP. The method MDA is provided in Table 7.

3.2.5.6. Plutonium-238

^{238}Pu has a half-life of 87.7 years and decays to ^{234}U by alpha particle emission. Because of the difference in half-lives between ^{238}Pu and ^{234}U, there is no significant in-growth of radioactive daughters. The energies and intensities of the two primary alpha particles emitted are

5456 keV	28.3%
5499 keV	71.6%

^{238}Pu emits the following low-energy photons that may be useful for radiobioassay

11.6 keV	0.196%
13.6 keV	3.97%
15.4 keV	0.114%
17.1 keV	5.57%
20.3 keV	1.28%
43.5 keV	0.0389%

RPDP's standard analysis for ^{238}U is alpha spectrometry of urine. The method MDA is provided in Table 7. Wound counting analysis has an MDA of 8.89E-3 uCi.

3.2.5.7. Uranium-238

^{238}U decays with a half-life of 4.468E9 years to ^{234}Th by alpha particle emission. ^{234}Th decays by beta particle emission and with a half-life of 24 days and grows into equilibrium with the ^{238}U. The energies and intensities of the two primary alpha particles emitted are:

4149 keV	22.9%
4198 keV	76.8%

^{238}U emits the following low-energy photons that may be useful for radiobioassay:

11.1 keV	0.141%
13.0 keV	2.96%
14.5 keV	0.0922%
16.1 keV	4.47%
19.1 keV	1.02%

In addition, ^{234}Th emits the following photons:

11.4 keV	0.178%
13.3 keV	3.66%
16.5 keV	4.7%
19.8 keV	1.23%
63.3 keV	3.81%
92.3 keV	2.73%
92.8 keV	2.69%
113.0 keV	0.242%

RPDP's standard analysis for ^{238}U is alpha spectrometry of urine. The method MDA is provided in Table 7.

3.2.5.8. Uranium-235

^{235}U decays with a half-life of 7.038E8 years to ^{231}Th by alpha particle emission. The ^{231}Th does not emit any photons useful for radiobioassay. ^{231}Th decays by beta particle emission with a half-life of 26 hours and grows rapidly into equilibrium with the ^{235}U. The energies and intensities of the two primary alpha particles emitted are:

4366 keV	17.6%
4398 keV	56.0%

^{235}U emits the following low-energy photons that may be useful for radiobioassay:

144 keV	10.5%
186 keV	54.0%

^{235}U is analyzed via alpha spectrometry of urine. The method MDA is provided in Table 7.

3.2.5.9. Uranium-234

^{234}U decays with a half-life of 2.445E5 years to ^{230}Th by alpha particle emission. Because of the half-life of ^{230}Th, there is no significant in-growth of radioactive daughters. The energies and intensities of the two primary alpha particles emitted are:

4721 keV	27.4%
4773 keV	72.3%

^{234}U emits the following low-energy photons that may be useful for radiobioassay:

11.1 keV	0.17%
13.0 keV	3.56%
14.5 keV	0.113%
16.1 keV	5.44%
19.1 keV	1.25%
53.2 keV	0.118%

RPDP's standard analysis for ^{234}U is alpha spectrometry of urine. The method MDA is provided in Table 7.

3.2.5.10. Mixtures of Uranium Isotopes

Enriched uranium is quantified with the 185 keV photon of ^{235}U; whereas, natural and depleted uranium are quantified with the 63 keV photon of ^{234}Th, the short-lived daughter of ^{238}U. Note that ^{234}U and ^{236}U are not easily quantified by chest counts because they lack high-energy, high-intensity photons. Thus, whichever photon is used to quantify uranium in the chest, the isotopic composition of the uranium must be known in order to calculate the total uranium content. Uranium seldom exists at the Savannah River Site as pure isotopes, i.e., we seldom encounter pure ^{238}U or ^{234}U. Rather, uranium typically exists as mixtures of isotopes like those listed[ix] below

- Highly enriched uranium (HEU) is used in applications like nuclear fuel.
- Depleted uranium (DU) is used for shielding, target material for production of higher Z nuclides, and various experiments.
- Natural uranium (U-Nat) is, as the name implies, the isotopic mixture of uranium observed in nature. Deviations from this activity mixture (with enrichment in ^{234}U) are quite common for U-Nat that has leached from rocks.

Known uranium source terms at SNL include HEU, DU, and U-Nat. The activity fractions of these uranium mixtures are presented in
Table 5.

[ix] The DU and NU data are from DOE Standard, *Guide of Good Practices for Occupational Radiological Protection in Uranium Facilities* DOE-STD-1136-2009; the HEU and RU data are from SRS sources.

Table 5, Activity fractions of uranium mixtures

	HEU	DU	U-Nat
^{238}U	0.0000	0.9015	0.4818
^{236}U	0.0000	0.0000	0.0000
^{235}U	0.0194	0.0145	0.0225
^{234}U	0.9806	0.0840	0.4957

MDAs are determined using the tracer radionuclide and the activity fraction of the tracer known to be present in the mixture. Using HEU as an example, ^{234}U is the tracer and the MDA is equal to the stand-alone ^{234}U MDA divided by the activity fraction of ^{234}U known to be present in HEU, or

(0.05 pCi/L) / 0.9806 = 0.051 pCi HEU/L

Uranium is analyzed via alpha spectrometry of urine. The method MDAs are provided in Table 6.

Table 6, MDAs for Uranium Mixtures

	Urine Radiobioassay Tracer	Urine MDA (pCi/L)
HEU	^{234}U	0.051
DU	^{238}U	0.055
U-Nat	^{234}U	0.101

3.2.5.11. Cesium-137

^{137}Cs decays with a half-life of 30 years by beta particle emission, with 94.6% of the decays producing the radioactive daughter Ba-137m, which has a half-life of 2.6 minutes. ^{137}Cs does not emit any photons useful for radiobioassay, but Ba-137m emits one 662 keV photon of 90% intensity that is useful for radiobioassay purposes. Thus, ^{137}Cs appears to emit one 662 keV photon of 84.4% intensity, taking into account the fraction of ^{137}Cs decaying to ^{137m}Ba.

^{137}Cs is measured *in vivo* by whole body counting based on the 662 keV photon, where the ^{137}Cs is assumed to be in equilibrium with the ^{137m}Ba. The method MDA is provided in Table 7.

3.2.5.12. Strontium-90

^{90}Sr decays with a half-life of 28.6 years by beta particle emission, to the radioactive daughter ^{90}Y, which has a half-life of 64 hours. Neither nuclide emits photons that are useful for radiobioassay. ^{90}Sr emits beta particles with a maximum energy of 546 keV and ^{90}Y emits beta particles with a maximum energy of 2283 keV. Gas-flow proportional counting is used to quantify ^{90}Sr/Sr-total in urine as the typical method for analysis. The method MDA is provided in Table 7.

3.2.5.13. Cobalt-60

^{60}Co decays with a half-life of 5.3 years by beta particle emission. The energy and intensity of photons emitted by ^{60}Co that may be useful for radiobioassay are:

1173 keV	100.0%
1332 keV	100.0%

^{60}Co is measured *in vivo* by whole body counting based on the 1332 keV photon. The method MDA is provided in Table 7.

3.2.5.14. Hydrogen-3

^{3}H (tritium) decays with a half-life of 12.3 years by beta particle emission. Liquid scintillation counting of urine is used to quantify tritium in urine via the 18.6 keV (max) beta particles emitted. The method MDA is provided in Table 7.

3.2.5.15. MDA Summary

A summary of the MDAs for the analytical methods used in the internal program are summarized in Table 7:

Table 7, MDA Summary for In Vivo and In Vitro Radiobioassay

Radionuclide	Method	Lab	Urine MDC (pCi/L)	Whole Body MDA (nCi)	Fecal MDA (pCi/sample)
^{241}Am	Alpha Spec	Offsite	0.05	N/A	0.05
^{241}Pu	Alpha Spec	Offsite	3.0	N/A	3.0
^{240}Pu	Alpha Spec	Offsite	0.05	N/A	0.05
^{239}Pu	Alpha Spec	Offsite	0.05	N/A	0.05
^{238}Pu	Alpha Spec	Offsite	0.05	N/A	0.05
6% Pu	Alpha Spec	Offsite	0.05	N/A	0.05
12% Pu	Alpha Spec	Offsite	0.05	N/A	0.05
^{238}U	Alpha Spec	Offsite	0.05	N/A	0.05
^{235}U	Alpha Spec	Offsite	0.05	N/A	0.05
^{234}U	Alpha Spec	Offsite	0.05	N/A	0.05
HEU	Alpha Spec	Offsite	0.051	N/A	N/A
DU	Alpha Spec	Offsite	0.055	N/A	N/A
U-nat	Alpha Spec	Offsite	0.101	N/A	N/A
^{137}Cs	Gamma Spec	RPSD	N/A	9.6	N/A
^{90}Sr/^{90}Y	Beta Counting	Offsite	N/A	N/A	5.0
^{60}Co	Gama Spec	RPSD	N/A	6.4	N/A
^{3}H	Direct LSC	RPSD	1000	N/A	N/A

N/A = not applicable, method not used

4. MINIMUM DETECTIBLE DOSE

4.1. Design Objectives for Internal Dosimetry Monitoring

A method of describing the ability of a given internal dosimetry program to detect dose is needed to judge whether or not a program meets the design objectives for radiobioassay programs in terms of effective dose. The term Minimum Detectable Dose (MDD) is used to describe the ability of an internal dosimetry program[20] to detect dose. MDD, discussed in detail in this section, is the CED that a radiobioassay program will detect 95% of the time.

The Minimum Detectable Intake (MDI) is the amount of radioactivity taken into the human body that has a 95% probability of being detected by a radiobioassay program. The CED delivered by the MDI is the Minimum Detectable Dose (MDD). The MDI and MDD are used to measure the efficacy of radiobioassay programs and should not be used for any other purpose. Specifically, the MDD should not be used to speculate whether or not a dose could have been associated with the < DL result for a particular radiobioassay.

In this section, the conceptual basis for MDD is presented and the MDDs are calculated for radionuclides included in the current SNL source term.

4.1.1. Minimum Detectable Dose for Routine Radiobioassay

The MDI for routine radiobioassay is the intake that will generate one of the following

- Urinary excretion such that a urine sample result is above the DL
- A body content[x] at least equal to the MDA of the whole body counter at the time of measurement

Notice that the MDI for urinary excretion follows the rules of engagement for assigning intakes discussed in Section 0. In the following section, the MDI will be specified only following a single acute inhalation intake of 5.0 µm AMAD particle size. The MDI(t) is evaluated as follows:

Equation 1, MDI(t) Calculation

$$MDI(t) = \frac{MDA}{e_u(t)}$$

The excretion fractions, $e_u(t)$, are calculated with the biokinetic models and methods discussed in Section 0, which are derived using commercial software such as (but not limited to) Integrated Modules for Bioassay Assessment (IMBA) with ICRP-based parameters, or taken from peer-

[x] Referred to as a body burden in the past

reviewed, published tables found in health physics literature such as ICRP 119[xi] or the Potter IRFs.[xii] We have a 95% probability of detecting MDI(t) because the MDAs from Section 3 are used. The MDD(t) is simply the product of the MDI(t) and the Dose Conversion Factor (DCF):

Equation 2, MDD(t) Calculation

$$MDD(t) = MDI(t) \cdot DCF$$

The DCF[21] or dose conversion factor is the effective dose coefficient for inhalation [$e_{inh}(50)$] or ingestion [$e_{ing}(50)$] found in ICRP 68, ICRP 119, and other ICRP 60+-based publications. DCFs of interest in the RPDP source term are listed in Section 6.17. By default, all MDDs were calculated assuming a single acute inhalation with ICRP 60-based DCFs for 5 µAMAD particles.

When discussing the MDD for a radiobioassay program it is convenient to have a single value. This provides correlation between the MDD for a particular analysis at a specific time interval, (MDD(t)), to the performance objective.(s) The urine, fecal, and whole body counter MDDs for the SNL source term are presented by nuclide or mixture and solubility type in Table 9, and Table 11 through Table 21.

Note that the time associated with routine sampling is 1-180 days for those radionuclides analyzed by urine radiobioassay and 1-365 days for those capable of being detected by whole body counting. Using Type M 6% weapons grade plutonium as an example, the MDD for a 6-month sample of Type M 6% plutonium to three significant digits is 1.34 rem CED. This MDD may be used to conclude that a biannual urine radiobioassay program for Type M 6% weapons grade plutonium can, by itself, meet the regulatory requirement of detecting an intake that would deliver a CED of 5 rem, and cannot reliably meet the performance objective of 100 mrem CED.

MDDs for routine fecal radiobioassay are not presented here because routine fecal radiobioassay is not performed at SNL.

4.1.2. Minimum Detectable Dose for Special Radiobioassay

A special radiobioassay program is presumed to consist of one or more of the following: a urine sample, a fecal sample, or a whole-body count. All are assumed to be performed within 7 days (168 hours) following a known or suspected intake of radioactive material.

The urine sample is assumed to have the concentration expected at 24 hours post-intake. Whole-body counts are assumed to have been performed 24 hours after the intake. The fecal sample is assumed to consist of the material excreted in the first 24 hours after the incident. For a single radiobioassay following a single intake, the MDI is simply given by:

[xi] ICRP Publication 119, *Compendium of Dose Coefficients based on ICRP Publication 60*, 2012.
[xii] *Intake Retention Fractions Developed from Models Used in the Determination of Dose Coefficients Developed for ICRP Publication 68 - Particulate Inhalation,* C.A Potter, Health Physics 83(5):594-789; 2002

Equation 3, MDI Calculation

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

where the IRF(t) is the intake retention fraction selected for the times given above. The MDD is calculated by multiplying the MDI by the appropriate DCF. The MDAs for routine radiobioassay are used for calculating the MDDs for special radiobioassay. Also, the MDAs for fecal radiobioassay are assumed to be the same as the corresponding MDAs for urine radiobioassay.

The MDD for special sampling is achieved only if sample(s) are collected within the expected 7-day period. For americium and plutonium Type S materials in particular, the only means of meeting the 100 mrem performance objective is an aggressive special sampling program (i.e. within 7 days of intake). This is best illustrated using Type S 6% weapons grade plutonium:

- The 7-day urine MDD for Type S 6% weapons grade plutonium is 0.297 rem.
- The 7-day fecal MDD for the same mixture is 0.001 rem.

From these MDDs it is concluded that:

- A 7-day urine sample can reliably meet the 5 rem regulatory requirement but cannot meet the performance objective of 100 mrem CED.
- A 7-day fecal can reliably meet both the 5 rem CED regulatory requirement and the 100 mrem CED performance objective.
- **In the event of a known or suspected intake, a special fecal sample must be collected within 7 days to bring the detection capability below 100 mrem.**

The MDDs for special radiobioassay are presented Table 9, and Table 11, through Table 21 in units of rem to three decimal places. This corresponds to the level of significance of recorded internal doses at SNL. Any MDD less than 0.001 rem CED is listed as 0.001 rem CED.

4.2. Calculated Minimum Detectible Dose for Personal Air Samples (PAS)

RPDP uses intake assessment as the principal means for most dose evaluations, with internal doses calculated based on estimated intake. The intake is estimated using available data, preferably *in vitro* or *in vivo* bioassay measurements. As discussed in Section 3.1, exposure time to air concentrations may also be used to supplement and/or indicate need for special bioassay measurements. For a PAS filter measurement collected in the individual's breathing zone, the MDD is given by:

Equation 4, MDD(t) Calculation

$$MDD(t) = \frac{MDA(t) \cdot DCF}{IRF(t)}$$

where MDA(t) is the MDA for analysis of the PAS filter. For personnel doses associated with PAS measurements, the intake retention fraction is the ratio of the PAS flow rate to Reference Man's breathing rate in liters per minute (lpm), or:

Equation 5, IRF(t) Calculation

$$IRF(t) = \frac{PAS\_Rate}{20}$$

The MDD(t) for PAS then becomes:

Equation 6, MDD(t) Refined Calculation

$$MDD(t) = \frac{MDA(t) \cdot DCF}{PAS\_Rate} \times 20$$

Using an inhalation intake of Type S ^{239}Pu as an example, the *a priori* MDAs for a PAS filters for a typical 4-hour work evolution are 4E-7 µCi/filter gross alpha 3E-6 µCi as counted on an iMATIC.[xiii] Using DCFs for 5 µAMAD particulates from ICRP Publication 68, *a priori* MDDs for nuclides of interest at SNL are presented in Table 8:

Table 8, A Priori PAS MDDs (rem)

	***A priori* MDA (uCi)**	**ICRP 68 DCF (Sv/Bq)**	**Type**	**IRF**	**MDD**
Sr-90	3E-06	7.70E-08	S	0.2	**0.001**
U-234	4E-07	6.8E-06	S	0.2	**0.001**
U-235	4E-07	6.1E-06	S	0.2	**0.001**
U-238	4E-07	5.7E-06	S	0.2	**0.001**
Pu-238	4E-07	1.1E-05	S	0.2	**0.001**
Pu-239/240	4.00E-07	8.3E-06	S	0.2	**0.001**
Pu-241	3.00E-06	8.4E-08	S	0.2	**0.001**
Am-241	4.00E-07	2.7E-05	M	0.2	**0.001**

The necessity for use of PAS is evaluated by RPDP in conjunction with workplace airborne radioactivity evaluations in RPO-06-15. Ideally this is performed prior to initiating work so that

[xiii] MDA and Relative Precision Evaluation by Analysis/Instrument for Calendar Year 2012, memo from RPSD Lab dated 6/14/2013.

PAS measurements may be collected throughout a work campaign and used to trigger bioassay campaigns. Rules regarding the use of PAS results in dose evaluations are discussed in Chapter 0.

4.3. Calculated In Vivo and In Vitro MDDs

A special radiobioassay program implemented within 24 hours of an intake is capable of meeting the performance objective of 100 mrem CED for an intake of any radioactive material. If implemented within a week post-intake, the same radiobioassay program is capable of demonstrating compliance with the federal limit of 5 rem CED.

The capabilities of the routine radiobioassay programs may be broken down into three categories:

- Routine program capable of meeting the 100 mrem CED performance objective
- Routine program incapable of meeting the performance objective but capable of demonstrating compliance with the 5 rem CED federal limit
- Routine program incapable of demonstrating compliance with the federal limit

At SNL, the only materials that fall into the third category are insoluble (Type S) plutonium and americium. This means that for these materials, a routine radiobioassay program alone cannot be used to demonstrate compliance with the federal limit[xiv]. **Special efforts must be made to identify intakes of insoluble (Type S) plutonium and americium as they happen and implement a special radiobioassay program no more than 7 days (168 hours) following the event.** If initiated under these specific conditions, a special fecal sampling program is fully capable of demonstrating compliance with the federal limit and the 100 mrem performance objective.

[xiv] Recall that at SRS the routine radiobioassay program is not used to demonstrate compliance with the federal limit.

4.3.1. Americium-241

Table 9, Americium-241 MDDs (rem)

Type M			
	Urine	Feces	Whole Body
7 days	0.870	0.001	N/A
180 days	0.457	N/A	N/A

Type S			
	Urine	Feces	Whole Body
1 day	0.530	0.001	N/A
7 days	1.390	0.001	N/A
180 days	3.700	N/A	N/A

From these MDDs it is concluded that:

- A biannual urinalysis program for Type M Am-241is not capable of meeting the 100 mrem CED performance objective but is capable of demonstrating compliance with the 5 rem CED federal limit.

- A biannual urinalysis program for Type S Am-241is not capable of meeting either the federal limit or the performance objective, unless special sampling was conducted within 24 hours of a known or suspected intake.

- **Collecting a fecal sample within 7 days following a known or suspected intake is required to demonstrate compliance with the 100 mrem performance objective.**

4.3.2. Plutonium

The plutonium mixtures for 6% Pu and 12% Pu as given in the Plutonium Good Practices Manual[22] are stated in Table 10.

Table 10, Activity Fractions for Mixtures of 6% and 12% Plutonium

	6% Pu Activity	12% Pu Activity
^{241}Pu	0.8367	0.9712
^{240}Pu	0.0278	0.0045
^{239}Pu	0.1181	0.0030
^{238}Pu	0.0174	0.0212

Thus, for 6% Pu, there are approximately 5 nCi of ^{241}Pu for every nCi of alpha emitting plutonium (αPu). For 12% Pu, the ^{241}Pu/ α Pu ratio is 34:1.

Immediately after separation ^{241}Am begins to grow in from the ^{241}Pu present in the material. Because of the relatively low urinalysis MDA of ^{241}Am in comparison to the MDA for pure plutonium, the ^{241}Am acts as a tracer for the plutonium.

An αPu/^{241}Am ratio of 10:1 will be assumed here for reference purposes. This ratio is attained after approximately 2 years for freshly produced 12% Pu and 19 years for freshly produced 6% Pu. By "freshly produced" it is meant that the plutonium was just made by irradiation of a target material in a reactor. As the plutonium ages, there is less of the ^{241}Pu present relative to the longer lived α Pu isotopes. This means that when aged plutonium is separated from the ^{241}Am it will take longer to attain the 10:1αPu/^{241}Am ratio or the 10:1 ratio may not be attainable at all.

For the MDDs given in Table 11 and

Table 12, both 6% Pu and 12% Pu refer to the isotopic mixtures assuming a 10:1 αPu/^{241}Am ratio which apply to both 6% and 12% Pu.

Table 11, MDDs for 6% Weapons Grade Plutonium Mixtures on Alpha Spec (rem)

Type M			
	Urine	Feces	Whole Body
1 day	0.031	0.001	N/A
7 days	0.297	0.001	N/A
180 days	1.344	N/A	N/A

Type S			
	Urine	Feces	Whole Body
1 day	0.796	0.001	N/A
7 days	5.988	0.001	N/A
180 days	11.674	N/A	N/A

From these 6% plutonium MDDs it is concluded that:

- A biannual urinalysis program for Type M 6% plutonium is not capable of meeting the 100 mrem CED performance objective but is capable of demonstrating compliance with the 5 rem CED federal limit.

- A biannual urinalysis program for Type S Am-241 is not capable of meeting either the federal limit or the performance objective, unless special sampling were conducted within 24 hours of a known or suspected intake.

- **Collecting a fecal sample within 7 days following a known or suspected intake is required to demonstrate compliance with the performance objective.**

Table 12, MDDs for 12% Fuels Grade Plutonium (rem)

Type M			
	Urine	Feces	Whole Body
1 day	0.154	0.001	N/A
7 days	1.460	0.016	N/A
180 days	6.610	N/A	N/A

Type S			
	Urine	Feces	Whole Body
1 day	3.970	0.001	
7 days	29.800	0.004	N/A
180 days	55.370	N/A	N/A

From these 12% plutonium MDDs it is concluded that:

- A biannual urinalysis program for Type S and M 12% plutonium is not capable of demonstrating compliance with either the 5 rem CED federal limit or the 100 mrem CED performance objective.

- **Collecting a fecal sample within 7 days following a known or suspected intake is required to demonstrate compliance with the performance objective.**

4.3.2.1. Plutonium-239 TIMS Urinalysis

Table 13, MDDs for TIMS Urinalysis of Pu-239 Alone (rem)

Type M			
	Urine	Feces	Whole Body
7 days	0.001	N/A	N/A
180 days	N/A	N/A	N/A

From these MDDs it is concluded that:

- **Collecting a plutonium urine sample for TIMS analysis within 7 days following a known or suspected intake is required to demonstrate compliance with both the 5 rem CED federal limit and the 100 mrem CED performance objective.**

4.3.3. Uranium

The MDDs for Type F solubility of all three isotopic mixtures defined in Table 5 and discussed in Section 3 are less than 10 mrem for radiobioassay so they will not be presented here.

Table 14, MDDs for Uranium Mixtures (rem)

Type M			
	180 days (urine)	7 days (urine)	Feces
DU	0.005	0.001	N/A
HEU	0.006	0.001	N/A
U-nat	0.011	0.001	N/A

Type S			
	180 days (urine)	7 days (urine)	Feces
DU	0.362	0.056	N/A
HEU	0.383	0.066	N/A
U-nat	0.383	0.066	N/A

From these MDDs it is concluded that:

- A biannual urinalysis program for Type M and S uranium is fully capable of meeting the 5 rem CED federal limit.
- A biannual urinalysis program for Type M uranium is fully capable of meeting the 100 mrem CED performance objective.
- **Collecting a urine sample for Type S uranium urinalysis within 7 days following a known or suspected intake is required to demonstrate compliance with the performance objective.**

Note that the MDD for urine radiobioassay presented here assumes no interference from non-occupational sources of uranium. In practice, the MDD for urine radiobioassay can be significantly higher because of the presence of naturally occurring uranium in the urine. Special efforts should be made to distinguish background uranium values to prevent the assignment of occupational dose from environmental sources.

4.3.4. Cesium-137

Table 15, MDDs for Cs-137 by Whole Body Counting (rem)

Type F	
180 days	0.002
365 days	0.006

From these MDDs it is concluded that:

- An annual whole body counting program for Type F cesium 137 is fully capable of meeting both the 5 rem CED federal limit and the 100 mrem CED performance objective.

4.3.5. Strontium-90

Table 16, MDDs for Sr-90 by Beta Counting (rem)

	Type F	Type S
7 days (feces)	0.001	0.001
7 days (urine)	0.001	0.016
180 days (urine)	0.010	0.454

From these MDDs it is concluded that:

- A biannual urinalysis program beta counting for Type M strontium is fully capable of meeting both the 5 rem federal limit and the 100 mrem CED performance objective.

- A biannual urinalysis program beta counting for Type S strontium is capable of meeting the federal limit but not the performance objective.

- **Collecting a fecal sample within 7 days following a known or suspected intake is required to demonstrate compliance with the 100 mrem performance objective.**

4.3.6. Cobalt-60

Table 17, MDDs for Co-60 by Whole Body Counting (rem)

	Type M	Type S
180 days	0.008	0.012
365 days	0.016	0.016

From these MDDs it is concluded that:

- An annual whole body counting program for Type M or S cobalt 60 is fully capable of meeting both the 5 rem CED federal limit and the 100 mrem CED performance objective.

4.3.7. Tritium

For tritium MDD calculation purposes, several assumptions are made:

- A 14-day sampling frequency for the duration of the tritium work

- Baseline tritium urine samples are submitted prior to performing radiological work

- The MDDs for stable metal tritide (SMT) and organically bound tritium (OBT) are calculated based on an annual frequency.

- The MDD presented for the SMT assumes a 5 µm AMAD and Type M dissolution in the lungs.

Table 18, MDDs for H-3 by Direct Liquid Scintillation Counting (rem)

	Type M SMT	OBT	HTO
1 day	0.001	0.001	0.001
14 days	N/A	N/A	0.005
365 days	0.037	0.016	N/A

From these MDDs it is concluded that:

- A biweekly urinalysis program for HTO is fully capable of meeting both the 5 rem CED federal limit and the 100 mrem CED performance objective.

- An annual urinalysis program for OBT or Type M SMT is fully capable of meeting the federal limit and the performance objective.

4.4. Summary Tables

Table 19, MDDs for Type F materials (rem)

	180 days Urine	365 days WBC	7 days Urine	1 day WBC
^{90}Sr	0.010		0.001	
^{137}Cs		0.006		0.001
U-Nat	0.001		0.001	
DU	0.001		0.001	
HEU	0.001		0.001	

Table 20, MDDs for Type M materials (rem)

	180 days Urine	365 Days WBC	7 days Urine	1 day Urine	7 days WBC	7 days Fecal
^{60}Co		0.016			0.001	
U-Nat	0.011		0.001			
DU	0.005		0.001			
HEU	0.006		0.001			
^{239}Pu (TIMS)			0.001	0.001		
6% Pu	0.344		0.297	0.031		0.001
12% Pu	0.613		0.460	0.154		0.004
^{241}Am	0.457		0.870	0.013		0.001

Table 21, MDDs for Type S materials (rem)

	180 days Urine	365 days WBC	7 days Urine	1 day Urine	7 days WBC	7 days Feces
^{60}Co		0.016			0.001	
^{90}Sr	0.454		0.016	0.002		0.001
U-Nat	0.383		0.056	0.008		
DU	0.362		0.056	0.008		
HEU	0.383		0.066	0.009		
6% Pu	1.674		0.988	0.796		0.001
12% Pu	5.366		9.831	3.970		0.016
^{241}Am	3.700		1.390	0.530		0.001

5. BIOASSAY ASSIGNMENT

5.1. Radiobioassay Programs

There are two distinct types of radiobioassay programs at SNL:

- The routine (confirmatory) radiobioassay program is designed to verify the accuracy of workplace monitoring data that indicates workers have not been internally exposed to radioactive materials. This program is considered the final quality control check of the engineered and administrative controls used to prevent occupational exposure to radioactive material.

- The special (for-cause) radiobioassay program is designed to assess the dose delivered by inadvertent intakes of radioactive materials that are likely to deliver a committed effective dose (CED) of more than 100 mrem during a calendar year.

5.1.1. Worker Categories

The need for a worker to be on a radiobioassay program is tied to the probability that the worker will receive intake(s) that will deliver in excess of 100 mrem CED in a calendar year. For practical purposes, the probability of a worker exceeding the 100 mrem monitoring level is broken down into three categories:

- Likely

- Reasonable potential

- No potential

Federal rule 10CFR835.402(c) *requires* internal dosimetry monitoring for workers who are "likely under typical conditions" to exceed 100 mrem CED in a calendar year. The word "likely" suggests a probability or eventuality that can reasonably be expected. "Typical conditions" are those conditions that are anticipated for a particular operation[xv]. Thus, the rule as interpreted by the SNL internal dosimetry program is interpreted as meaning that workers intentionally exposed to radioactive materials that will deliver in excess of 100 mrem are *required* to be monitored (which includes radiobioassay)[xvi]. Workers at SNL are not intentionally exposed to airborne radioactive contamination, which means that the 100 mrem level is exceeded only as a result of unplanned releases. Thus, only the special radiobioassay program will fall under the

[xv] In practice, "typical conditions" may be considered to be conditions that do not exceed the suspension guides of the Radiological Technical Work Document (RTWD).
[xvi] The *Guide of Good Practices for Occupational Radiological Protection in Plutonium Facilities*, DOE-STD-1128-2008, Section 5.3.2, states that "… no typical plutonium worker is likely to have intakes of 100 mrem CED or more."

10CFR835.402(c) requirement. However, in most cases of prescribed special radiobioassay programs at SNL, the worker is still not considered likely to exceed the 100 mrem monitoring level.

Workers who are unlikely to exceed the monitoring level but, because of the nature of their work have a "reasonable potential" for exposure to airborne radioactive materials, may be placed on a routine radiobioassay program[xvii]. The routine radiobioassay program, which is not mandatory under 10CFR835.402(c), is prescribed for various members of the workforce (MOW) on a case-by-case basis through discussion and agreement between dosimetry, the radiation protection line support team, the management of the member of the workforce, and the worker. Members of the workforce who routinely work with large quantities of unencapsulated radioactive material and use respiratory protection devices for radiological purposes[xviii] are the most likely candidates for routine radiobioassay programs. A routine radiobioassay program for such workers is a valuable measure of the effectiveness of procedural and engineered controls and, more importantly, to substantiate the *a priori* judgment about likelihood of an intake[xix]. This last point is an important one. Because of uncertainties in determining exactly who is likely to exceed 100 mrem we must extend the routine radiobioassay program beyond those who we think are "likely" to those who we think have "reasonable potential."

Finally, there are workers who seldom encounter significant unencapsulated sources and do not use respiratory protection devices for radiological purposes. These workers are deemed to have no reasonable potential for an intake and should not participate in routine radiobioassay programs. Note that these workers may still be monitored by workplace air sampling and contamination surveys, and a special radiobioassay program would be prescribed to these workers should there be any reasonable indication of an internal exposure.

5.1.2. Respiratory Protection and Likelihood

As part of the prospective determination of the likelihood of the occurrence of an intake, credit may be taken for respiratory protection if[23]:

- The work activities are well planned and controlled
- There is timely and accurate monitoring of the workplace
- There is a demonstrable history of implementing effective work controls

[xvii] The *Standard for Internal Dosimetry*, DOE-STD-1121-2008, section 5.1, states that "… the need for an internal dosimetry program is linked more to the potential for intake than the likelihood of intake."
[xviii] DOE Enforcement Guidance Supplement EGS 99-02 states that credit may be claimed for respiratory protection when determining likelihood of exceeding the monitoring level.
[xix] The *Guide of Good Practices for Occupational Radiological Protection in Plutonium Facilities*, DOE-STD-1128-2008, section 5.3.2, states that "Although no one should be considered likely to have intakes resulting in 100 mrem CED, some workers are at significantly higher risk for incurring an intake than others and should be on routine bioassay"

- The respiratory protection program meets the requirements of 29CFR1910.134[24]

Because work at SNL meets all four criteria, credit is taken for respiratory protection in the determination of likelihood. This means that for a worker in an airborne radioactivity area (ARA) wearing respiratory protection it is deemed unlikely for the dose to exceed 100 mrem CED under typical conditions.

5.1.3. Identifying Program Source Terms

A routine radiobioassay program can detect previously unidentified intakes of radionuclides that a worker had a "reasonable potential" to be exposed to. This is in contrast to a special radiobioassay program which is designed to assess intakes of specific radionuclides to which a worker has been exposed. The key points are:

- A special radiobioassay program is initiated in response to a real, measurable source term (e. g., radioactive material on an air filter paper) that the worker was exposed to.

- A routine radiobioassay program is initiated in response to a "hypothetical" source term. That is, a particular mixture of radionuclides that might typically be expected to be present at a particular location.

The easiest facility to identify the analytes of interest for routine radiobioassay is a facility that has only processed one specific type of radioactive material. The advantage here is that no matter how closely one looks at the facility in time and/or space, the source term is the same. At the other extreme is the facility that has always processed or handled many different types of radioactive materials over time. The difficulty here is that the source term may change in time and/or space.

Another issue with identifying the source term for a routine radiobioassay program is that the information available on mixtures of radioactive materials present in a facility will address macroscopic quantities such as kg or Ci. Information is seldom available on the dosimetrically important microscopic level of µg or nCi. Industry practices have shown that the composition of the source term at these two different magnitudes can be quite different. The result of this "low-level heterogeneity" is that what a worker is most likely to be exposed to, at the microscopic level, may not be what the process engineer thinks is in the facility, at the macroscopic level. This potential discrepancy is one of the main reasons why the source term identified for a routine radiobioassay program should never be used to specify a special radiobioassay program. Instead, the source term for the special radiobioassay program should always be specifically identified.

The specification of the source term for a routine radiobioassay program is largely a professional judgment because it is a prediction, based on limited information and data, of the mixture of radionuclides that a worker might be exposed to at some undetermined time in the future. On the other hand, because special radiobioassay is triggered by a specific event and there is usually a sample of the material available for isotopic speciation, there is less professional judgment needed to adequately specify the program.

If a radiobioassay program and the source term are not properly matched, an intake could go unidentified or evaluated improperly such that the assigned dose is incorrect. Thus, great care must be taken to match a special radiobioassay program to the correct source term because the radiobioassay program may be used to demonstrate compliance with 10CFR835. A mismatch between the special radiobioassay program and the appropriate source term can lead to violations of the federal rule with potential Price Anderson Amendment Act (PAAA) enforcement actions. Because a routine radiobioassay program is not used to demonstrate compliance with the federal rule and is prescribed only for workers who are not likely to have had an intake, the consequences of a mismatch between the routine radiobioassay program and the source term are not as dire as in the case of special radiobioassay. In short, a mismatched routine radiobioassay program is a misallocation of resources and a source of false confidence in the radiological protection program. As such, care should be taken to closely match the routine radiobioassay program to the source term.

5.1.4. Radiobioassay Program Performance Requirements

As addressed in Section 4, an internal dose monitoring program (including radiobioassay) required under 10CFR835.402(c) must be able to demonstrate compliance with the 5 rem Total Effective Dose (TED) limit and the 50 rem Total Equivalent Dose (TEqD) limit. Note that these limits are the sum of the Effective Dose and the 50-year Committed Effective Dose (CED) and 50-year Committed Equivalent Dose (CEqD), respectively. In summary:

$$TED = ED + CED$$

$$TEqD = ED + CEqD$$

A radiobioassay program that is not required by 10CFR835.402(c) does not have any regulatory performance requirements. It is the policy of the Radiation Protection department at SNL to maintain personnel doses As Low As Reasonably Achievable (ALARA)[xx], and as such, no deliberate intakes are authorized in order to aid in complying with federal exposure regulations. The SNL administrative control level (ACL) for occupational exposure to Radiological Workers is 250 mrem[xxi25]. In practice, this ACL applies to dose received from external sources since internal doses are not expected to occur. This means that the radiobioassay program must be able to identify intakes of radioactive material that, in a calendar year, would deliver 4.75 rem CED in order to ensure that workers are not exceeding the 5 rem federal limit.

As discussed in Section 4, for certain radionuclides (especially actinides), currently available radiobioassay technology may not permit us to demonstrate compliance with the 100 mrem TED dose limit for minors and members of the general public. Therefore, these individuals should not be permitted to enter areas where there is an increased potential for intakes of radioactive materials.

[xx] MN471016 *Radiological Protection Procedures Manual*, Section *7.1.2* SNL ALARA Policy Statement
[xxi] MN471016 *Radiological Protection Procedures Manual, Section 1.4.2.2.1.1 Radiological Worker Dose Limits*

5.1.5. Performance Objectives for Radiobioassay Programs

A radiobioassay program is considered to be adequate if it can show compliance with the federal limits, which is, in essence, an upper design criterion. Thus, the lower design criterion is given as the performance objective.

Because federal rules require workers to be monitored at 100 mrem in a year, it would be both logical and desirable to have the radiobioassay programs be able to detect such a dose. Therefore, the performance objective for radiobioassay programs is to detect intakes of radioactive material in a year that will deliver a CED of 100 mrem from each "independent source. " The special radiobioassay program can meet the performance objective for intakes of all materials. However, it is not feasible for the routine radiobioassay program to meet the performance objective for intakes of most insoluble actinides. The inability to meet the performance objective is referred to as a "technology shortfall."

The performance objective is deemed to be an objective that, once met, indicates that further improvement in the monitoring capability of the bioassay program will usually be unwarranted. The performance objective is not meant to be a detection limit that must be met at any cost[xxii]. Although the DOE has not provided explicit guidance on this issue, this approach is consistent with that given by the USNRC[26].

5.1.5.1. Independent Sources

The performance objective cited above is considered to apply to each independent source of radioactive material in the workplace[xxiii]. For example, if a worker handles tritium in one building and plutonium in another (i.e., they are independent sources), the monitoring program for the worker should be able to detect 100 mrem from intakes of tritium and 100 mrem from intakes of plutonium. If the sources are not independent, the performance objective applies to the mixture. Note that the federal dose limits apply to the sum of all occupational sources of radiation and radioactive material, thus, the independent source rule does not apply.

5.1.6. Technology Shortfall

For routine radiobioassay, the performance objective may not be achievable in a convenient, cost-effective manner with currently available technology (especially for insoluble actinides). If there is a technology shortfall in the routine radiobioassay program, DOE[27,28] recommends:

- Implementation of an aggressive special radiobioassay program

[xxii] It is debatable whether or not the concept of technology shortfall even applies to a radiobioassay program (like the SNL routine radiobioassay program) that is not required by the Federal Rule.
[xxiii] The concept of an independent source appeared in early versions of the Implementation Guide (see *Department of Energy Internal Dosimetry Implementation Guide*, G-10 CFR 835/C1, December 1993) but is not discussed in recent versions.

- Use of supplementary air monitoring (including PAS when appropriate)
- Use of "best practice" radiobioassay monitoring methods

Thus, a special radiobioassay program is frequently prescribed for workers who, though unlikely to have exceeded the monitoring level in a specific instance, have an abnormally elevated potential for an intake. Note that for whatever reason a special radiobioassay program is prescribed, it is considered mandatory due to the recourse for the technology shortfall of the routine radiobioassay program.

5.1.7. MDD versus Frequencies

Numerous factors go into the design of a radiobioassay program including:

- MDD
- Monitoring the buildup of long half-life material in the body
- Desired accuracy in dose assessments
- Cost
- Time away from the job for affected worker(s)
- Controlling worker exposures
- Past practices and experiences
- Worker acceptance

Note that all of these factors can lead to the establishment of a radiobioassay frequency that is different than the frequency indicated by MDD analysis alone.

5.2. Routine Radiobioassay

Routine radiobioassay is performed at a prescribed interval for workers who have reasonable potential for exposure to radioactive materials. As previously discussed, this program is used as a final quality assurance check on engineered and procedural controls and is not used to meet the requirements of 10CFR835.402(c). The site-level procedure governing routine radiobioassay is RPDP-03-01, "Bioassay Assignment and Termination[29]" The philosophy and general implementation of the routine radiobioassay program is discussed here.

5.2.1. Routine Radiobioassay Programs

There are three different types of routine radiobioassay:

- Verification radiobioassay verifies that workers deemed not likely to be exposed to radioactive material are, indeed, not being exposed to radioactive material based on the results of radiobioassay measurements. This type of radiobioassay is what we typically mean when we refer to "routine" radiobioassay, and is sometimes referred to as "confirmatory" bioassay.

- Baseline radiobioassay establishes the monitoring status of workers who are beginning work at SNL that may require routine radiobioassay.

- Closeout radiobioassay establishes the radiological status of workers who are ending work at SNL that required routine radiobioassay.

The term "monitoring status" refers to how much internal dose, if any, a worker has received from radioactive materials deposited within the body and the ability to detect additional intakes given what is already present in the body.

5.2.1.1. Verification Radiobioassay

As previously discussed, workers are not intentionally exposed to radioactive materials at SNL. This policy of maintaining doses ALARA[30] makes it unlikely that a worker will exceed a dose of 100 mrem CED. However, routine radiobioassay programs are prescribed for members of the workforce if they have access to a source of radioactive material that, in a plausible release scenario, could result in doses exceeding 100 mrem CED (i.e., they have "reasonable potential"). In this situation, routine radiobioassay is the final quality control check used to verify the adequacy of

- Engineered and administrative controls

- Workplace monitoring

- The determination of "likelihood"

Last but not least, the routine radiobioassay program is "… a kind of safety net to identify intakes which might have gone undetected by workplace monitoring". [31] It must be emphasized that workplace monitoring alone cannot be used to demonstrate compliance with federal dose limits for workers.

5.2.1.2. Baseline Radiobioassay

Workers who are assigned to work at SNL which could result in an intake and have not performed similar work previously at SNL are required to complete a baseline radiobioassay.

Workers requesting a dosimeter are screened to determine if they were on radiobioassay programs at other sites or have had an intake of radioactive material. Based on the results of this screening, workers may be asked to submit a baseline radiobioassay sample. If active SNL employees performed radiological work at locations other than SNL, they should also be screened prior to returning to radiological work onsite to see if a baseline radiobioassay is required.

5.2.1.3. Closeout Radiobioassay

Once a MOW performs activities under an RTWD that requires radiobioassay, the prescribed program must be completed. The program is normally completed within 30 days of the end of the work activity, or upon leaving the work activity, whichever happens sooner. If the worker terminates employment at SNL, the prescribed program must be completed at the time of termination announcement.

5.2.2. Specification of Routine Radiobioassay Programs

To specify a routine radiobioassay program the following questions must be answered:

- What are the radionuclide(s) of concern
- What are the types and capabilities of the radiobioassay that will be performed
- How frequently will the radiobioassay be performed
- Who will participate in the program.

There is little specific guidance from regulatory or standards organizations on the implementation and administration of a routine radiobioassay program. Therefore, considerable professional judgment must be used to interpret the intent of the available guidance, incorporate this with expectations of industry good practice, and assign workers to the appropriate routine radiobioassay programs.

The following sections give a general discussion of the thought processes that go into specifying a routine radiobioassay program.

5.2.2.1. Radionuclides of Concern

Although there may be many radionuclides present in a source term, typically only a few have the potential for delivering significant doses. The important radionuclides are usually quite obvious: uranium in uranium facilities, plutonium in plutonium facilities, and tritium in tritium facilities, for example. To balance the need to properly match the routine radiobioassay program and the source term with the effort and expense required to do so, routine radiobioassay programs are typically designed to monitor for the most *dosimetrically significant* radionuclides that are likely to be present. These radionuclides are referred to as “radionuclides of concern.”

Also, some radionuclides can be important not because of the dose they deliver but because of the relative ease of detection, making them able to be used as tracers for the radionuclides that actually deliver the dose. ^{241}Am in a plutonium facility being a good example of this.

Radionuclides of concern are determined in the following manner[32]: All radionuclides in a work area to which workers could be exposed are identified from waste certification records, contamination surveys, elevated air sample results, safety analysis reports, technical reports, the open literature, personal interviews, etc. The radionuclides in the area that deliver a cumulative dose fraction[xxiv] of more than 90% are deemed to be the radionuclides of concern for the RPDP and are to be considered for inclusion on the Radiological Technical Work Document (RTWD) by the Radiation Protection department (RP). All other radionuclides may be ignored for the purposes of bioassay unless they are suitable for use as a tracer.

The determination of the radionuclides of concern is very much a professional judgment because it is seldom feasible to perform a complete real-time source term analysis at multiple locations. The most effective practical approach is to have radiological engineers in the facility where radiological work will be happening who are knowledgeable of the past and current operations in the facility and are cognizant of, and react to, significant changes in the source term.

5.2.2.2. Types of Routine Radiobioassay

The following types of radiobioassay may be used in the routine radiobioassay program:

- Urine radiobioassay, to measure radioactive material cleared from the blood and systemic organs
- Feces radiobioassay, to measure radioactive material cleared from the gastrointestinal and respiratory tracts
- Whole body counting, to measure radioactive material in the body

It is important to note that each type of radiobioassay provides independent and unique information, i.e., they are complementary. Urine radiobioassay is typically used for materials that cannot be readily detected by whole body counting or chest counting, such as tritium and ^{90}Sr. Whole body counting is used for radionuclides that emit penetrating photon radiation such as most gamma-emitting fission and activation products. Feces bioassay is useful primarily for materials that can be retained in the lungs and emit low-energy photons, such as the actinides. Routine feces radiobioassay is currently not implemented at SNL, but it can be an important part of the special radiobioassay program following suspected inhalation of non-tritium radionuclides.

[xxiv] The largest dose fractions, that, when combined, are equal to or greater than 90%.

5.2.2.3. Routine Radiobioassay Frequency

Ideally, the frequency for routine bioassay is selected so that the MDD for the procedure is less than the performance objective. For tritiated water, where the objective is easily met, other considerations such as desired accuracy in the dose assessment are considered. The ICRP[33] recommends that the 2σ error on doses less than 1 rem not exceed a factor of 2, which is an error of 100%. Analyses of simulated chronic intakes of tritiated water show that a monthly sample frequency will produce an error of less than 100%. [34] At the other extreme are the insoluble actinides, for which it is not currently possible to meet the performance objective. In this instance, historical precedent (what has worked in the past), practicality, and making an effort to demonstrate compliance with regulatory limits become dominant factors.

Using a variety of sampling frequencies can lead to problems with the implementation of multiple RTWDs. For example, if a quarterly plutonium frequency is specified on the RTWD and a worker is already on an annual plutonium program elsewhere, do the annual samples count as part of the quarterly samples, or do the quarterly samples have to remain a separate program? To minimize these sorts of problems, a bi-annual frequency was adopted for all *in vitro* radiobioassay and an annual frequency was adopted for all *in vivo* radiobioassay, with the exception of tritium.

The requirement for workers to pick up and submit routine urine samples at a central collection facility (Building 869, Room B13), was implemented to ensure samples were properly identified, had adequate volume, and to ensure the fidelity of the chain of custody for each sample.

Two potential drawbacks of a bi-annual in vitro routine sampling frequency are that:

- Should an intake be detected by routine radiobioassay, one must review 6 full months of work history (rather than 3 months or 1 month) in an effort to identify when the intake occurred
- The MDD for routine radiobioassay will usually increase as the sampling frequency decreases.

In response to the first potential drawback, the radiological controls and work controls programs are designed to prevent the release of radioactive contamination and identify potential exposures of workers to said contamination. Such potential exposures are promptly evaluated with special radiobioassay programs. The routine radiobioassay program for actinides is a final quality control check of the programs designed to prevent and identify exposures – the routine radiobioassay program is not intended to, nor can it, compensate for deficiencies in these other programs. In response to the second potential drawback, the MDD is indeed higher for a bi-annual sampling frequency versus quarterly or monthly. However, this sampling frequency did not make any routine radiobioassay program have an unacceptable MDD.

5.2.2.4. Selection of Workers

Only workers identified as a Radiological Worker in the RPDP program should be considered for participation in routine radiobioassay programs. While Radiological Worker 2 training tells a worker how to access areas where there is unencapsulated radiological material, many MOW at

SNL have been identified as having taken Radiological Worker 2 training without actually performing Radiological work.

Routine radiobioassay programs are prescribed *prospectively*, meaning that a worker is determined to be in a radiobioassay program before the anticipated work is performed. This determination is mainly made based on the expected job title and / or duties of the individual. The determination can also include consideration of the RTWDs the individual is expected to sign in on as part of their duties.

This begs the question of how the radiobioassay requirements are identified for specific tasks on RTWDs. In practice, radiobioassay is highly recommended whenever respiratory protection devices are used for radiological purposes. Respiratory protection is required in both:

- Areas posted as a high contamination area[xxv] (HCA) or an airborne radioactivity area[xxvi] (ARA)

- Whenever hands-on work having the potential for releasing contamination is performed in the presence of a significant source term.

Note that only a HCA or ARA is considered to have sufficient unencapsulated radioactive material to present a reasonable potential for an intake (and hence require routine radiobioassay). Contamination areas (CA) are not considered to present a reasonable potential for an intake. These judgments are based on consideration of the source term typically present in a CA and operational experience. These are general guidelines. A routine radiobioassay program can be required for a particular task in a CA by the radiological technical work document for that task. For example, consider the situation where a glovebox containing a significant source is located in a room posted as a CA. Workers using the glovebox may be required to participate in a routine radiobioassay program (because of the potential for a release) whereas workers passing through the CA may not[xxvii].

5.2.2.5. Monitoring Period for Radiobioassay

A worker who is required to participate in a routine radiobioassay program must submit a sample and/or receive *in vivo* counts as requested. Workers are informed by electronic mail of

[xxv] Greater than 100 times 10 CFR 835 Appendix D contamination levels
[xxvi] Greater than 0.1 DAC airborne radioactivity
[xxvii] Should a CA be found to contain unanticipated levels of contamination, a special radiobioassay program can be instituted after the fact.

radiobioassay(s) that are due. These notification letters are sent out as the sample kits become ready to pick up. This means that the monitoring period is from the sample stop time (completion date) of one radiobioassay to the next sample start time, not from the date of one notification letter to the next.

5.2.3. Radiobioassay Programs for Special Tritium Compounds

Tritium in the urine is typically considered to be the result of intakes of tritiated water (HTO or tritium oxide). However, in the recent past, considerable interest has been shown in the dosimetry of special tritium compounds (STCs)[35]. STCs consist of tritium incorporated into organic materials (organically bound tritium or OBT) and tritium incorporated into metals (stable metal tritides or SMTs). Although intakes of OBT give slightly more dose and generate slightly lower excretion rates than equivalent intakes of HTO, they do not pose any significant detection or interpretation problems. On the other hand, SMTs are particulates that can be rather insoluble (absorption Type M and S) and can be relatively difficult to detect and interpret, especially in the presence of tritium oxide. Locations that meet one or both of the following criteria[36] should be characterized in terms of the fraction of total tritium present as an SMT:

- Areas posted as an ARA/HCA for total tritium
- Areas with an unencapsulated tritium source in excess of 83 Ci.

Standard Sandia protocol is a 100-mL baseline single void urine sample followed by 14-day routine samples for the duration of the tritium campaign. Special sampling involves one or more single-void samples collected (ideally) within 2-4 hours of completing tritium work.

5.2.4. Routine Bioassay for Declared Pregnant Workers

When a worker has an intake of radioactive material, a fraction of the intake which is termed the uptake, is transferred to the blood. If the worker is a pregnant woman, part of the uptake can be transferred to the unborn child. This uptake by the unborn child can deliver dose to the child while it is still in utero. The DOE has set a dose equivalent limit of 500 mrem to the unborn child (which is called an embryo/fetus) for the nine month gestation period. It is important to note that this limit is defined in terms of *dose equivalent* (not effective dose) received during the *gestation period* (not the 50 years following the intake).

Assuming an acute inhalation at time of conception, the following materials and intake quantities would produce a gestation period dose of 500 mrem. These intakes were calculated using the dose coefficients from ICRP 88. [37]

- Type M ^{238}Pu: 1 µCi
- Type F ^{137}Cs: 50 µCi
- Type M ^{90}Sr: 50 µCi

Several conclusions can be drawn from these values:

- Depending on the absorption type, the inhalation intake that produces the uptake can be from 2 to 20 times greater than the uptake. For example, an intake of Type S ^{238}Pu would be much larger than Type M ^{238}Pu to deliver the same CED.

- Maternal intakes of unprecedented magnitude would be required to produce uptakes that could deliver 500 mrem to the embryo/fetus. For example, a 12.75 µCi intake of Type S ^{238}Pu corresponds to a 50-year CED of 500 rem. The largest intake that has occurred at SNL in the past decade delivered a CED of approximately 22 mrem.

- Currently available radiobioassay techniques can easily detect a very small maternal uptake, orders of magnitude below the limit.

From these observations, the following can be concluded:

- Under normal operating conditions at SNL, any internal dose to the embryo/fetus will be negligible. The only plausible way the 500 mrem limit could be exceeded would be by exposure to external sources of radiation.

- Current routine radiobioassay programs designed to monitor radiation workers are adequate to monitor pregnant workers.

5.2.5. Routine Radiobioassay Programs for Soluble Uranium

Soluble uranium poses a chemical toxicity hazard, primarily to the kidneys. DOE does not specify chemical toxicity based exposure limits for soluble uranium, so the Occupational Safety and Health Administration (OSHA) limit[xxviii] of 0.05 mg/m^3 is recommended[38]. Assuming a specific activity (SA) of 4x10^{-7} Ci/g for depleted uranium (DU)[xxix], this translates (with appropriate unit conversion) into an air concentration of

Equation 7, Air concentration example

$$\left(\frac{0.05\,\mathrm{mg}}{\mathrm{m}^3}\right)\cdot \mathrm{SA} = 2\times 10^{-11}\,\frac{\mu\mathrm{Ci}}{\mathrm{cm}^3}$$

which is less than the default DAC for uranium used at SNL. This means that if exposures to uranium are controlled on the basis of radiological toxicity (the DAC), then exposures to uranium will be automatically controlled on the basis of chemical toxicity. An 8 DAC-hr exposure to depleted uranium produces an intake of 1704 dpm (1920 µg), which is easily detected by the special urine radiobioassay program that will ensue.

[xxviii] 8-hour time weighted mean air concentration
[xxix] DU gives the lowest air concentration and is therefore the most restrictive case.

In the uranium industry, high sampling frequencies[xxx] have historically been used to assess and control the chemical toxicity of soluble uranium. However, at SNL a semi-annual sampling frequency is used for all MOW working with uranium. The discussion given above and the fact that SNL uranium workers have a low potential for significant exposures provides the technical basis for the semi-annual sampling frequency. This approach is consistent with the guidance given in the Uranium Good Practice Manual[39].

5.3. Special Radiobioassay

The special (for-cause) radiobioassay program meets the performance requirements for all radioactive materials addressed in this manual.

Special radiobioassay programs are prescribed for workers who are deemed likely to incur an intake of radioactive material that will deliver a CED in excess of 100 mrem. These radiobioassay programs are therefore considered to fall under the regulatory requirements of Federal Rule 10CFR835.402(c). In practice, special radiobioassay programs are initiated:

- In response to a radiological incident to determine if an intake has occurred
- As follow-up to a known intake in order to quantify the intake and monitor the excretion and/or retention in the worker
- As follow-up to a positive routine radiobioassay

The special radiobioassay programs for radioactive materials are outlined in RPDP-08-01, "Responding to Radiological Incidents" [40] and RPDP-08-02, "Determining Internal Doses."[41] Special radiobioassay programs for assessment of intakes are prescribed by the internal dosimetrist. The technical basis for these programs will be discussed in this section. The ability of special radiobioassay to detect intakes is discussed in Chapter 4.

5.3.1. Special Radiobioassay for Incident Response (Non-Tritium)

The quantity of radioactive material in the body and in the excreta is typically highest during the first few days following an intake. This means that radiobioassay performed promptly after an intake will permit detection of the smallest possible intakes. Experience has shown that prompt assessment leads to the most accurate assessment of the intake. For these reasons, special radiobioassay is prescribed whenever there is an increased risk of an intake having occurred. In general, the risk of an intake is elevated whenever containment of the radioactive material is lost, i.e., the material becomes airborne, is on workplace surfaces, or is on the skin or modesty clothing of worker(s). Once containment is lost, the decision to prescribe special radiobioassay depends on the following:

[xxx] Daily or weekly

- Quantity of material released
- Duration of exposure
- Use of respiratory protection and anti-contamination measures.

The criteria for initiating a special radiobioassay program must be developed through experience with the relevant processes and materials and are specific to individual facilities. If a process changes or new radiation protection guidelines and regulations are implemented, these criteria may also change.

5.3.1.1. When Special Radiobioassay Programs Are Required

A special radiobioassay program is required when a worker is suspected to have had an intake of radioactive material. Experience at SNL and among the internal dosimetry industry has shown that an intake should be suspected if any of the following occurs:

- A worker is exposed to airborne radioactivity in excess of 8 DAC-hr in a day or the indicated air concentration could greatly underestimate the exposure to the worker. The values for air concentration and exposure assume the assigned protection factor for any respiratory protection in use will be applied.
- Contamination is measured on single-layer protective clothing in excess of 1,000 dpm alpha per 100 cm2 or 10,000 dpm beta-gamma per 100 cm2 if respiratory protection is not used.
- Contamination is measured on the inner layer of multiple-layer protective clothing in excess of 10,000 dpm alpha per 100 cm2 or 100,000 dpm beta-gamma per 100 cm2 if respiratory protection is used.
- An unplanned release of radioactive material produces contamination on accessible surfaces in excess of 1,500 dpm alpha per 100 cm2 or 15,000 dpm beta-gamma per 100 cm2 if respiratory protection is not used
- Any detectable personal contamination is measured on the hair, face, neck, chest, arms, or hands, or anywhere else on the body in excess of 500 dpm alpha per 100 cm2 or 5000 dpm beta-gamma per 100 cm2 if respiratory protection is not used
- A worker incurs a contaminated wound.

There are many factors that could modify the guidelines given above. For example, if the contamination on a shoe cover is identified as a hot particle or if the contamination event is immediately identified and corrective actions taken, the incident may not require a special radiobioassay program. Thus, it is important to note that the above are truly guidelines, and that the internal dosimetrist responding to an incident will prescribe the appropriate radiobioassay program based on experience and professional judgment.

5.3.1.2. Types of Special Radiobioassay Programs

A special radiobioassay program may consist of one or more of the types of analyses described below:

True 24-hour urine samples are preferable for all special radiobioassay programs except for those assessing intakes of tritium oxide. A 24-hour urine sample consists of all the urine excreted by an individual over a 24-hour period. Volumes observed for 24-hour urine samples can range from 300 mL to over 6 L. Note that there is no minimal acceptable volume for a properly collected 24-hour urine sample. In other words, a 300 mL 24-hour urine sample is acceptable if it was collected over a 24 hour period.

Fecal samples may be collected following a suspected intake of radioactive material and are required for many actinides due to technology shortfall with urinalysis. Because gastrointestinal transit times vary considerably from person to person, it is difficult to suggest a minimal acceptable mass on a single sample, but 50-100 grams wet mass is a typical expected range. There is a time delay in excretion of ingested material after an intake that is caused by transit of the material through the GI tract. This delay can range from hours to days, depending on the individual. Feces samples collected too soon after an intake can be misleading because they may contain no activity. For this reason, fecal samples are requested to be collected 24 hours post event. Fecal samples should be promptly screened (direct x-ray and photon counting after ashing) for activity and additional samples collected if significant activity is detected. Incidents with significant potential for an intake exceeding occupational exposure limits, especially those involving plutonium, may warrant the collection of all fecal excretions for the first 7 days.

Whole body counts, as appropriate, should be performed as soon as possible after a suspected intake of a photon emitter. If there is no external contamination on the worker, the counts will provide valuable information on the initial deposition of material in the body. It is vital that the worker be free of external contamination during in-vivo radiobioassay. This is accomplished by means of a portal monitor stationed at the entrance to the RPSD whole body count room. Small quantities of external contamination may be misinterpreted as large quantities of internal contamination. A whole body count is adequate for detecting gamma-emitting fission and activation products. Urine radiobioassay is required for pure beta-emitters such as ^{90}Sr and tritium. Attention should be given to the possibility of actinides being present if fission products are detectable.

It is recommended that a routine urine sample that is positive be followed up with a single 24-hour urine sample.

5.3.1.3. Composition of Radioactive Materials

There are two things to remember about the composition of radioactive materials at SNL

- Most radionuclides exist as mixtures

- The importance of each radionuclide in a mixture depends on the technical perspective of whom you ask.

For example, a worker may be exposed to highly enriched uranium reported by the facility as being essentially pure (>98%) ^{235}U. The material may be pure ^{235}U by mass but it is essentially pure ^{234}U by activity. The radionuclides to which workers are exposed during an incident should be identified by analysis of the contaminant(s) if feasible. Note that the relative quantities of various radionuclides in excreta will most likely be quite different than the quantities in the inhaled material due to the action of human metabolism on the material.

The method for determination of the radionuclides of interest for routine radiobioassay programs does not apply to special radiobioassay. When a special radiobioassay program is prescribed, representative samples of the contamination to which the individuals were exposed should be analyzed and the isotopic fractions of the mixture determined. The methods discussed for routine radiobioassay should only be used when no other information is available.

5.3.2. Special Radiobioassay for Incident Response (Tritium)

Special radiobioassay for tritium is performed:

- If a process-related liquid is found on a worker's outermost garment

- If an intake of greater than 210 µCi (which gives a peak concentration of 5 µCi/L of tritium in the urine) may have occurred.

Urine samples should be collected no sooner than 90 minutes after the suspected intake. This allows time for the tritiated water to equilibrate with urine already in the bladder.

5.3.3. Long Term Follow-Up Radiobioassay

Long term follow-up radiobioassay programs are recommended if any future MOW who continue to be active employees have continued measurable excretion or retention of radioactive materials.

The purpose of this program would be to ensure that predicted excretion and retention are consistent with observed excretion and retention at extended times after an intake. Significant differences between the predicted and observed values should trigger a re-evaluation of the intake to determine the cause of the difference. Workers with significant body burdens of actinides should be invited to participate in the US Transuranium and Uranium Registries (USTUR)[42,xxxi].

[xxxi] See http://www.ustur.wsu.edu for more information.

5.3.4. Survey of Medical Procedures

The Medical Department at SNL uses information proved by both the Radiation Protection Line Support Teams involved in an incident as well as the Dosimetry Department to determine when and if medical procedures such as chelation are necessary. At the point that medical intervention is determined to be necessary, the Radiation Emergency Assistance Center / Training Site (REAC/TS) [xxxii] will be contacted for further guidance.

In general industry practice, medical intervention is considered if an intake has a potential to deliver a CED in excess of 2 rem, but the professional judgment of the physician plays a dominant role in determining the actions to be taken for any particular case.

[xxxii] See http://orise.orau.gov/reacts/ for more information.

6. INTAKE AND DOSE CALCULATION

An intake is the quantity of radioactive material that passes through the nares, mouth, or skin. Once a worker's intake has been estimated, it is usually a straightforward task to calculate dose. For practically all occupational exposures, intakes cannot be measured directly, but must be inferred from other measurements such as radiobioassay and air samples. This inference requires a biokinetic model that can relate intake to radiobioassay data such as urine and feces excretion or air concentration. Intake calculations can become quite complex, but have the advantage of at least attempting to account for all radioactive material that enters the body, regardless if it can be measured directly. However, before an intake can be evaluated, appropriate radiobioassay data must be collected and interpreted, including the decision as to whether an intake has actually occurred.

6.1. Special Radiobioassay for Intake Evaluation

The evaluation of an intake is basically an iterative process of postulating and verifying models. For example, a contamination event triggers a conservative intake estimate based on the available data and standard assumptions. Thus, we have postulated a model that describes all relevant aspects of the intake such as time of intake, the mode of intake, biokinetic models, etc. Key assumptions in the model are then identified and radiobioassay data are collected to justify more accurate (usually less conservative) assumptions in the model.

This process is repeated until a high degree of confidence in determination of the magnitude of intake is attained, which could mean either of the following:

- Additional information will not justify any significant changes in the model
- A conservative estimate of the dose is so small that continued refinements are unwarranted.

Thus, the uncertain aspects of the model may dictate the data needed and the "experiments" (i.e., the type and frequency of radiobioassay) that should be performed. The internal dosimetrist performing the evaluation takes these points into consideration when prescribing a radiobioassay program for evaluating an intake.

6.2. Air Sampling

Personal air samplers (PAS) are used to measure exposure to airborne radioactive material. The results of which are used to estimate intake from the exposure, bypassing problems with interpreting retention and excretion.

Work environments with airborne radioactivity potential are evaluated using NUREG-1400[8b] methodology in accordance with RP procedure[43]. If the evaluation determines that the calculated threshold coefficient (V_i) is greater than 2, (indicating a CED of 100 mrem or more), personal air sampling is required to be performed and reported to RPDP. PAS monitoring of applicable

individuals is conducted by RP operations personnel and reported to RPDP. Any individual that exceeds action levels prescribed for that specific work (often 8 DAC-hours/day) is prescribed special bioassay follow-up based on the radionuclides indicated by the PAS result. Note that if a PAS is used, and an intake of radioactive materials is measured, the intake will be assigned to the person in accordance with the discussion in this section.

6.3. Radiobioassay

Radiobioassay may be categorized either *in vivo* or *in vitro*. In an *in vivo* radiobioassay, the material in the body is quantified by the radiation it emits, which greatly simplifies the calculation of intakes by allowing the following to be accomplished:

- Determining some information as to the location of material in the body as a function of time
- Placing an upper limit on material that could be in the body and not be detected.

In vivo radiobioassay can be complicated by the presence of:

- External contamination on the worker
- Materials that emit non-penetrating photons
- Materials that emit photons that interfere with the measurement

When using *in vitro* radiobioassay, radioactive material in excreta from the body is quantified. The principle advantage of *in vitro* radiobioassay is that it allows evaluation of intakes of materials that are difficult or impossible to quantify using *in vivo* methods. However, what is in the body must be inferred by measuring what comes out of the body. This inference is usually more difficult and less accurate than direct measurement.

To evaluate radiobioassay data, certain interpretations must be made concerning how measurements relate to models. This section will discuss these interpretations for the following types of *in vivo* and *in vitro* radiobioassay:

- Whole body counting
- Incremental and non-incremental urine
- Accumulated and spot feces
- Nasal irrigation and smears
- Wound counts

6.3.1. Whole Body Counting

Whole body counts are performed with large NaI detectors in a standing geometry. The whole body counter measures the total activity in the body but gives little information on the distribution of material. The distribution is assumed to be that dictated by the biokinetic models. In practice, this assumption is satisfactory because the dose resulting from material quantified by whole body counting (usually fission and activation products) is typically quite small and the distribution of the material does not greatly influence the dose. The distribution of material in the body should be determined empirically for intakes that deliver CEDs that are a significant fraction of regulatory limits. External contamination fixed to the skin can cause erroneous whole body counts. The final whole body count, after all decontamination efforts are completed, is assumed to be valid unless there is defensible evidence that there is non-removable external contamination on the worker.

6.3.1.1. Cesium-137 in Workers from Non-SNL Sources

^{137}Cs is present in the environment from atmospheric weapons testing. The ^{137}Cs can work its way through the food chain and end up in humans, for example, through the consumption of wild game. This natural background can cause problems in interpreting whole body count data. The levels of ^{137}Cs typically observed in workers do not represent a significant dose alone, but, if the ^{137}Cs is used as a tracer, small quantities may represent relatively large doses. The following guidelines should be followed concerning ^{137}Cs[44]:

- If feasible, ^{137}Cs should not be used as a tracer in designing radiobioassay programs
- ^{137}Cs detected in the bodies of workers following a routine whole body count should be assumed to come from non-SNL sources if the worker routinely consumes wild game and the body content is less than 20 nCi. No further actions are required in this case.
- ^{137}Cs detected in the bodies of radiation workers from a special whole body count should be assumed to come from non-SNL sources if the worker routinely consumes wild game, the body content is consistent with the results of previous whole body counts and is less than 20 nCi. No further actions are required in this case.
- For all other cases where ^{137}Cs is detected in the body a special radiobioassay program, including analyses for ^{90}Sr and plutonium, should be initiated.

6.3.2. Urine Radiobioassay

Urine radiobioassay is so prevalent at SNL that it is often simply referred to as “bioassay.” The primary advantage of urine radiobioassay is that it allows the evaluation of intakes of materials that are difficult or impossible to quantify by *in vivo* counting. The primary disadvantage of urine radiobioassay is inferring what is in the body (the intake) based on the quantity of material being excreted in the urine over some period of time. This process is one step removed from estimating

the intake from what is in the body (*in vivo* radiobioassay) and hence can introduce additional errors into the final intake estimate.

The presence of radioactive material in the urine is usually considered proof that the material was in the systemic body[xxxiii] and thus is a good indicator of the soluble portion of an intake. On the other hand, this means that urine radiobioassay may not provide any information concerning the insoluble portion of an intake.

6.3.2.1. Time Interval of a Urine Sample

In vivo measurements are usually interpreted as instantaneous measurements at a point in time. For example, we assume the distribution of the radioactive material in the body does not change during the time it takes to perform the measurement. For this reason, *in vivo* retention functions that are used to evaluate *in vivo* data are expressed as the fraction of an intake expected to be present at a given point in time. Conversely, most *in vitro* measurements represent the collection of a sample over a time interval. The excretion functions used to evaluate the in-vitro data are, therefore, expressed in terms of the fraction of an intake excreted over a given time period.

Thus, to evaluate *in vitro* radiobioassay data, a time interval must be ascribed to each sample. There are several different methods commonly used today. The accuracies associated with each of these methods have not been documented. In order of preference, they are:

- Method A: The times for each void are recorded, and the interval calculated as the difference between the first and last void times. This method will always give the right answers, if done correctly, but in practice is seldom done correctly.

- Method B: The creatinine in a sample is measured and the time interval calculated assuming a constant creatinine output per day.

- Method C: The specific gravity of the sample is measured and the time interval calculated assuming a constant output rate of solids in the urine.

- Method D: The volume of the sample is measured, and the time interval is calculated assuming a constant volume of urine is excreted per day (1.4 liters per day is the typical value). This method is simple and easy, but is probably the least accurate of the techniques.

For best results, the bladder should be emptied at every void, and that is what is assumed. Methods A and D are most commonly used at SNL. Method B was evaluated at the Savannah River Test Site and found to be unsuitable for routine use[45,46,47].

Samples frequently span non-integral time periods, for example, from 1.1 days to 1.7 days, giving a time interval of 0.6 days ending at 1.7 days after the intake. In these cases, the excretion

[xxxiii] The systemic body excludes the lumen of the GI tract and lungs, which are technically outside of the body.

function can be tailored to the sample, or the sample can be tailored to the time period. The excretion function is readily adapted to the sample by looking at the expected excretion for the interval from day 1.1 to day 1.7. The problem is that each sample can represent different time intervals, which means that there is no expectation line with which empirical results can be compared.

The alternative is to somehow normalize the radiobioassay results to integral one-day values. For example, the result from day 1.1 to day 1.7 could be adjusted to give the result from day 1 to day 2. This method allows all the expectation and empirical results to be readily compared on the same graph. The disadvantage with this method is that additional uncertainty can be introduced into the radiobioassay result in the normalization procedure. Consider, for example, the following urine radiobioassay results:

Table 22, Normalizing Void Times

Urine Sample	Date of Void	Time of Void	Volume(ml)	Pu Conc (dpm/L)
1	May 9 May 10	0950 1900 0200	900	6.7
2	May 11	0305 1005 2108	950	6.7

If the time of the last void, prior to sample 1, was 0630 on May 9, which is assumed to be the time of intake, the activity in the first sample is:

Equation 8, First sample DPM

$$6.7\frac{dpm}{L} * 0.900\ L = 6.0\frac{dpm}{sample}$$

and in the second sample:

Equation 9, Second sample DPM

$$6.7\frac{dpm}{L} * 0.950\ L = 6.3\frac{dpm}{sample}$$

Table 23, Activity per sample (dpm)

Urine Sample	Start Time (days)	End Time (days)	Time Interval (days)	Activity (dpm per sample)
1	0.00	0.81	0.81	6.0
2	0.81	2.61	1.80	6.3

These results can be modeled as 6.0 dpm excreted from t = 0.00 day to t = 0.81 day and 6.3 dpm excreted from t = 0.81 day to t = 2.61 days. The alternative method is to normalize the results to integral 1 day intervals:

Equation 10, Excretion on Day 1

$$excretion\ on\ day\ 1 = \frac{6.0}{0.81} = 7.4\ dpm$$

and

Equation 11, Excretion on Day 2

$$excretion\ on\ day\ 2 = \frac{6.3}{1.80} = 3.5\ dpm$$

At SNL, the normalization method is preferred because it permits evaluation of all data graphically on one plot.

Urine samples that are adjusted according to their volume as described in method D are called non-incremental urine samples. For example, routine urine samples are usually considered to be non-incremental samples. At SNL the concentration of non-incremental urine samples is typically reported in units of dpm per liter. The urine output is assumed to be 1.4 liters per day; so, the concentration multiplied by 1.4 is the approximate daily excretion rate. This method can produce wide variations in the excretion rate and is used only if there is no information concerning the time interval a sample represents.

6.3.2.2. Urine radiobioassay for Uranium

Uranium in urine may be analyzed using a chemical property of uranium, in which case the results will be in units of mass. This is referred to as an elemental analysis for uranium. On the other hand, uranium may be analyzed using the radiation emitted by radioactive isotopes, in which case the results will be in units of activity. This is referred to as an isotopic analysis for uranium. Elemental analysis is typically used to evaluate chemical toxicity problems; whereas, isotopic analysis is typically used to assign occupational dose. For enriched uranium, most of the dose comes from ^{234}U. In any event, the dose per unit intake of ^{234}U, ^{235}U, ^{236}U, and ^{238}U are so close that the precise isotopic composition of uranium in the urine is not absolutely needed for dose calculations.

6.3.2.2.1. Background Uranium in Urine

The primary concern with uranium urine radiobioassay has always been differentiating occupational uranium from natural uranium. By activity, natural uranium has roughly a 1:1 ratio of ^{234}U: ^{238}U. Lacking access to data on the true background uranium levels in the urine of SNL workers, reasonable action levels for uranium results in urine were derived based on the blank background sample population. In order to be investigated as possibly from occupational sources, the uranium in urine had to exceed these action levels.

However, some urine bioassay results may still be from naturally occurring uranium. At this point, the best method to determine if it is from natural uranium or not is by examining the rations of ^{234}U, ^{235}U, and ^{238}U. Generally, naturally occurring uranium results have a $^{234}U/^{238}U$ ratio between 0.8 and 3.0.

In addition, if the detected isotope is ^{235}U or ^{236}U, ^{234}U must also be detected or else the result is considered to be a false positive, as discussed in Section 6.7. Once the uranium is considered to be detected, the following decision rules are applied to determine the origin of the uranium, assuming the worker is not actually working with natural uranium:

- If both ^{234}U and ^{238}U are detected and the $^{234}U/^{238}U$ activity ratio is between 0.8 and 3, the uranium is considered to be from a non-occupational source. Any material with a ratio outside of this range is considered to be from occupational sources.

- If only ^{238}U is detected, the uranium is considered to be from an occupational source only if the ratio of the ^{234}U D_L to the ^{238}U concentration is less than 0.8. The test fails if the ratio is greater than or equal to 0.8 and the material is assumed to be from a non-occupational source.

- If only ^{234}U is detected, the uranium is considered to be from an occupational source only if the ratio of the ^{234}U concentration to the ^{238}U D_L is greater than 3. The test fails if the ratio is less than or equal to 3 and the material is assumed to be from a non-occupational source.

If the uranium in the urine is considered to come from natural sources, the results are recorded and no further actions are taken. If the uranium is considered to come from occupational sources, appropriate actions are taken to confirm the intake, investigate the positive results, or assign the dose. These procedures will result in a MDD of less than 100 mrem CED for Type F and M uranium and less than 500 mrem CED for Type S uranium.

6.3.2.3. Total and Isotopic Urine Radiobioassay for Plutonium

Plutonium in urine is analyzed by alpha spectrometry, which gives results by isotope. Evaluations may need to be performed on mixes of total and isotopic plutonium urine data. This may be accomplished by summing the isotopic results to produce a total plutonium value, from which a total plutonium intake is calculated and then partitioned into isotopic plutonium intakes.

6.3.2.3.1. Plutonium as a Tracer for Neptunium

Trace quantities of ^{238}Pu are often associated with ^{237}Np because ^{237}Np is the target material used to produce ^{238}Pu. However, because of the large variability[48] in the Np/Pu ratio, a default mixture cannot be assumed. Because of this variability and the different metabolism of plutonium and neptunium, the absence of ^{238}Pu in excreta cannot be used to confirm the absence of ^{237}Np in excreta.

6.3.2.4. Contaminated Urine Samples

Urine samples could possibly become contaminated with non-occupational radioactive material. An example would be a tritium urine sample submitted by a worker who has recently been administered ^{99m}Tc for a cardiac stress test. In this case, ^{99m}Tc bleeds into the energy window set for tritium on the liquid scintillation counter. Only the chemists who analyze the urine samples can declare that a positive result is due to extraneous material and the interpretation must be documented. The internal dosimetrist will assume that the reported value is correct unless additional data can be collected to show that the result in question is an outlier.

6.3.2.5. Effect of Chelation on Urinary Excretion

Chelation therapy, with the zinc and calcium salts of DTPA (Diethylenetriamine pentaacetate), may be used at SNL after significant accidental intakes of transuranics to accelerate the removal of the material from the body. The Bioassay Lab must be aware of urine samples that have been submitted by a chelated worker as there is an additional clean-up step required in the preparation of the sample for analysis. If the clean-up steps are not performed the quantity of transuranic material in the sample may be underreported by up to 30%. Chelation therapy will complicate the evaluation of an intake because standard biokinetic models describing urinary excretion may not be applicable especially when modeling early urine data. In general, the effects of chelation on urinary excretion are assumed to subside completely within 100 days following cessation of the therapy.

6.3.3. Feces Radiobioassay

In contrast to a urine radiobioassay, feces radiobioassay can (and usually does) contain material that was never in the systemic body, i.e., insoluble material cleared from the upper respiratory tract and GI tract. Feces radiobioassay is thus complementary to urine radiobioassay. In practice, it is highly recommended that feces radiobioassay is always used in conjunction with a urine radiobioassay and never by itself.

6.3.3.1. Time Interval of Feces Radiobioassay

The material measured in a feces sample is the sum of excretion from the systemic body, translocation from the lungs, and unabsorbed ingested material accumulated over a certain time interval. This time interval can be difficult to specify because there can be considerable and variable transit time in the GI tract. To minimize this problem, feces samples for suspected large intakes should be collected over a time period that is long compared to the GI tract transit time. For example, collection of all feces over the first week then evaluated with an accumulated feces excretion model. This is particularly important in the first week following an intake. A single isolated feces sample should be assigned the time interval between voids, and if this time is unknown, an interval of 1 day is used. A single feces void should also never be collected immediately after (within 4 hours of) a contamination event because the GI transit time may cause a false negative value.

6.3.3.2. Systemic Excretion to the Feces

There is excretion of material from the liver to GI tract via the common bile duct. For plutonium, the quantity of material excreted in the bile is approximately the same as that excreted in the urine[49]. The excretion of plutonium in the bile may be ignored in the first week or so following an inhalation intake because it is typically small compared to the early translocation of material from the lungs. The impact of excretion via the bile should be determined for other time periods and materials.

6.3.3.3. Effect of Chelation on Feces Excretion

As discussed previously, chelation therapy immediately after an intake can greatly increase the urinary excretion rate of plutonium and other transuranics. Most of this increase is assumed to come from the chelation of free plutonium in the blood; that is, the increase in urinary excretion reflects unincorporated plutonium. Excretion of plutonium in the bile also increases but reflects the removal of plutonium that was incorporated in the liver[50]. As a rough approximation for an acute injection intake of plutonium, it is assumed that the feces excretion rate will be ~1/3 of the urinary excretion rate following chelation therapy.

6.3.3.4. Use of Tracers in Feces Radiobioassay

If a mixture of materials with different transport and solubility characteristics is inhaled, the composition of the mixture may change as it makes its way to the feces. Thus, the composition of material in the feces may not be representative of the composition of the material that was inhaled. The ratio of tracers to other materials should always be determined from samples of the inhaled material rather than excreted material.

6.3.3.5. Contaminated Feces Radiobioassay

Feces samples may be occasionally contaminated with extraneous radioactive material. Only the chemists who analyze the feces samples can declare that a positive result is due to extraneous material and the interpretation must be documented. The internal dosimetrists will assume that the reported value is correct unless additional data can be collected to show that the result in question is an outlier.

6.4. Wounds

The quantity of radioactive material that is deposited in a wound and the rate at which it is translocated to other parts of the body is the primary interest of the internal dosimetrist. These parameters are of primary importance for evaluating urine radiobioassay data. In the event of large intakes, evaluating feces radiobioassay data may be beneficial. Material excised from a wound should be quantified and isotopic ratios of the contaminants determined.

Most radioactive material in a wound will exhibit one or more of the following behaviors:

- Absorbed from the wound into the bloodstream
- Transported from the wound to the lymphatic system
- Retained in tissue around the wound
- Removed from the wound to the environment by natural causes (e. g. scab).

The dose to a wound site is seldom calculated as it is of limited value, especially for transuranic materials. Problems associated with quantifying the material in wounds are:

- The depth of deposition and self-absorption of radiation emitted by the material can create significant uncertainties in the counting efficiency of direct measurement, especially for low energy photon emissions
- If the location of the material is identified, Medical may elect to debride the wound and apply silver nitrate, which strongly absorbs low energy photons.

NCRP Report 156[51] was published in 2006 with the express purpose of providing biokinetic models for evaluation of contaminated wound intakes. These models were a collaborative effort between the NCRP and the ICRP in order to avoid redundancy between the two organizations. The primary driver for development of the guidance was contaminated wounds caused by radioactive depleted uranium fragments to military personnel during the 1991 Persian Gulf War. The general wound model is comprised of seven compartments of which five describe radionuclide behavior at the wound site, and two can receive radionuclides transported from the wound site. The general model is presented in Figure 1. The NCRP Report 156 wound models were incorporated into IMBA v4.1.18.

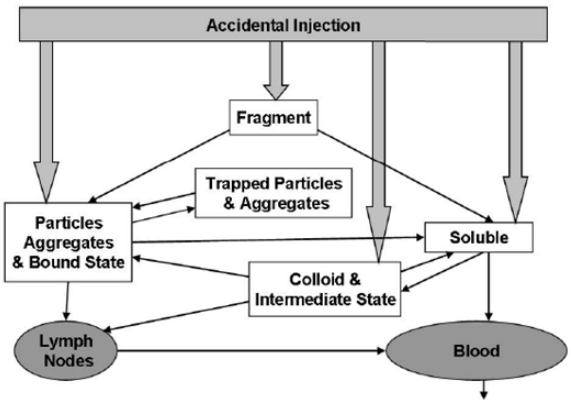

Figure 1, NCRP Report 156 general compartment model

6.5. Nasal Irrigation and Nasal Smears

The biokinetic models for the nasal region of the respiratory tract are, at best, very rough approximations of what is actually happening to material in the nasal region. For this reason,

nasal irrigation is viewed primarily as a therapeutic procedure. Material removed from the nasal region may be analyzed to determine the composition of the inhaled material. A nasal smear is a quick and simple technique used to detect the presence removable contamination in the anterior nose. The presence of such contamination in the nose can be a good indicator that the worker inhaled radioactive material. This is useful in determining the need for appropriate bioassay measurements consistent with 10CFR835.209. It is common health physics knowledge that the absence of removable contamination in the anterior nose does not prove that an intake did not occur[52]. In other words, the absence of proof is not proof of absence. Since nasal smear results represent activity not taken into the body, they cannot be used to assess or infer an internal dose. This is supported by a study[53] of historic intakes at the Savannah River Site showing that there is no correlation between the amount of radioactive material removed from the nasal region and the final CED assigned.

Nasal and saliva smears at SNL should only be implemented in extreme and dire circumstances though most DOE sites still employ these smears and in some cases[54] use the results for initial dose estimations.

6.6. Decision Rules for Radiobioassay Data

Once a radiobioassay is performed, four questions must be answered:

- Are the analytical results indicative of the presence of radioactive material in the body? For example, does a urine sample contain ^{238}U or does a whole body count indicate the presence of ^{137}Cs?

- If the analyte is detected, is it metabolized[xxxiv] material? For example, did the ^{238}U in the urine come from the worker or is it contamination that was never in the worker? Is the ^{137}Cs detected by a whole body count in the worker's body or is it external contamination?

- If the material is decided to have been metabolized, is it the result of an occupational exposure? In other words, did the ^{238}U and ^{137}Cs assimilated by the worker come from SNL or from some other non-occupational source?

- If the answer to all three questions is yes, it must be decided if the data are indicative of a "dosimetrically significant" intake. A dosimetrically significant intake is an intake that will deliver a dose that must be assigned and possibly investigated. For example, even if the ^{137}Cs detected by the whole body counter is actually in the worker's body and came from an occupational exposure, it may deliver such a low dose that no further actions are required. On the other hand, the ^{238}U in a urine sample may indicate a significant intake that warrants further follow-up and evaluation.

[xxxiv] Metabolized material is material that is or was in the body, as opposed to contamination outside the body.

For most radiobioassay results, the question must be answered as to whether or not the analyte is present because this answer will dictate the necessary follow-up actions. This question of detection is answered in a two-step process. In the first step, a statistical decision level (also called a critical level) is used to decide if the counts in an energy region of a sample spectrum are significantly greater than in the same region in a background spectrum. This process is discussed in numerous references[55,56,57]. The second step of the process involves a review of the spectrum by a cognizant technical function (CTF) to confirm the conclusions of the first step. For example, the application of a decision level to a whole body count spectrum may lead to the conclusion that no ^{60}Co is present, whereas a ^{60}Co photopeak may be clearly visible to the CTF. In such cases, the CTF may decide to overrule the initial decision and declare that ^{60}Co is present.

After determining analyte presence, two types of errors can be committed as shown below. If the analyte is present and the conclusion is incorrectly made that it is not, the result is a false negative. On the other hand, if the analyte is not present and the conclusion is incorrectly made that it is, the result is a false positive. These errors are illustrated in Table 24.

Table 24, Possible conclusions reached for a radiobioassay result

		Analytical Determination	
		Analyte	No Analyte
Reality	Analyte	Correct	Incorrect "false negative"
	No Analyte	Incorrect "false positive"	Correct

At SNL, the *in vitro* radiobioassay program for actinides attempts to achieve a false positive rate of approximately 1 to 2 percent. This means that for every 100 blank samples analyzed, 1 to 2 will be declared to contain analyte. The false negative rate for these samples in that case is approximately 5 percent when the sample contains analyte at the minimum detectable concentration. The false negative rate will increase as the level of the analyte decreases.

For a specific *in vitro* sample, it is very difficult to differentiate between metabolized and non-metabolized material. The decision is typically made by examining the results of blanks, reference standards, and other samples in a sample batch. If material is present in the blanks or a large fraction of the batch, the material is usually assumed to be non-metabolized. For *in vivo* counts, a differentiation can sometimes be made between external contamination and internally deposited material based on the presence of beta radiation on the skin. In any event, material still present after all decontamination efforts have been completed is assumed to be internally deposited.

6.6.1. Decision Rules for Intakes Detected by Radiobioassay Data

At this point it may have been concluded that an excreta sample or worker contains metabolized radioactive material from an occupational source. If radiobioassay did not have any systematic or

random errors, this result alone would indicate that an intake occurred. Likewise, it may have been concluded that the excreta sample or worker did not contain metabolized radioactive material from an occupational source. Once again, can two types of errors can be made, as shown in Table 25.

Table 25, Possible conclusions reached in an intake evaluation

		Conclusion	
		Intake	No Intake
Reality	Intake	Correct	Incorrect "missed intake"
	No Intake	Incorrect "phantom intake"	Correct

If an intake occurred and it was incorrectly concluded that it did not, an intake has been missed. If the intake did not occur and it was incorrectly concluded that it did, a "phantom[xxxv] intake" will be assigned. Ideally, we would like the doses associated with both phantom intakes and missed intakes to be small; specifically, less than 100 mrem CED. A dose of 100 mrem CED is important because it is the investigation level specified by DOE[58]. For materials such as tritiated water and most fission/activation products, the dose associated with missed and phantom intakes is less than 100 mrem CED. For actinides, the dose associated with phantom and missed intakes could be much greater than 100 mrem CED.

To keep the number of phantom intakes to a manageable level, certain rules are employed to decide if an intake has occurred. In the initial evaluation, an intake is assumed to have occurred in the following instances:

- Rule A: a single positive special radiobioassay measurement associated with a known incident
- Rule B: a positive routine radiobioassay measurement is followed by another positive radiobioassay measurement
- Rule C: a positive radiobioassay measurement is obtained and an appropriate routine radiobioassay measurement is not obtained.

Rule A means that approximately 2 out of every 100 workers involved in an incident will be falsely assigned an intake. The magnitude of a phantom intake due to a special radiobioassay will almost always be small because the sample or count was analyzed shortly after a known incident. Therefore, the dose erroneously assigned as a result of Rule A is typically much less than 100 mrem CED.

[xxxv] In this context "phantom" means something that does not exist.

Rule B means that in a routine radiobioassay program with a 2% false positive rate there is a chance of approximately $0.02^2 = 0.0004$ of a phantom intake. Thus, approximately 4 out of every 10,000 workers sampled on a routine radiobioassay program[xxxvi] will be assigned a phantom intake. The dose associated with these intakes would typically be less than 500 mrem CED because of the solubility class assumed and the fact that large intakes will produce highly significant radiobioassay data.

The phantom intake rate is easily approximated for these rules because the person did not have an intake. This means that the probability of a phantom intake is a function of the analytical process alone. However, the missed intake rate is much more difficult to calculate in the general case because it depends on the analytical process and the exact nature of the intake: the intake size, intake pattern, solubility class, aerosol size, and the times of radiobioassay relative to the time of intake. Thus, a good estimate of the frequency of missed intakes that deliver smaller doses is not known. The errors associated with missed intakes can be minimized by detecting intakes shortly after they occur (special radiobioassay) and not by routine radiobioassay.

Rule C is used primarily in the evaluation of historic radiobioassay data where the follow-up may not have met current standards and practices. This rule is invoked at the discretion of the internal dosimetrist.

Because the evaluation of low-level intakes of materials like plutonium frequently contains inconsistencies, the intake evaluation is considered by a peer Internal Dosimetrist before it is finalized and reported. The purpose of the peer is to review the overall intake assessment and ensure that it is reasonable and supported by the available data. It is important to note that once Rule A or B is invoked, no additional evidence is <u>required</u> to assign the intake. In other words, in the absence of any compelling information[xxxvii] to the contrary, the intake will be assigned.

6.7. Calculating Intake from Radiobioassay

After the decision that an occupational intake has occurred, the intake evaluation process begins by selecting a biokinetic model[59,60] which describes the intake, retention, and excretion of a radioactive material in an idealized Reference Man. This model specifies

- How material enters the body, how much is deposited, and the rate at which it leaves the deposition site
- The rate at which material will feed into the bloodstream and the gastrointestinal tract
- Where the material will reside in the systemic organs and for how long

[xxxvi] The reader is reminded that, as previously discussed, only workers with "reasonable potential" for exposure are monitored with routine radiobioassay.
[xxxvii] Once Rule A or B is invoked, the burden of proof shifts from proving that the intake has occurred to proving that it did not.

- Where the material will be excreted and at what rate

The biokinetic model is selected with the hope that it adequately describes what is happening to material in a real worker, e. g. John Doe for this example. Reference Man is given an intake of radioactive material, and expectation radiobioassay data are generated that match the types and times of John Doe's empirical radiobioassay data. Working on the assumption that biokinetics of the material in John Doe and Reference Man are the same, any difference in the expectation and empirical radiobioassay data may be attributed to the magnitude of the intake. The intake is therefore adjusted to produce the "best" match between the expectation and empirical data (exactly what "best" means will be discussed later).

In its simplest sense, the process just described is iterative radiobioassay evaluation. In this case, one parameter, the intake, was changed to make the expectation and empirical radiobioassay data match. Frequently, there are systematic differences between the expectation and empirical radiobioassay that cannot be reconciled by adjusting the intake alone. In these cases, the assumption that the biokinetics of Reference Man and John Doe are the same is considered incorrect and the biokinetic model is modified to obtain better agreement. Modifications can range from changing the particle size of inhaled material to changing half-lives in systemic compartments.

The problems with modifying biokinetic models are that several different modifications can cause the same effect in the expectation radiobioassay data and the empirical data from an occupational exposure may be woefully inadequate for adjusting parameters in a biokinetic model. The end result may be that, even though the expectation radiobioassay data and empirical radiobioassay data match, the wrong model was selected and the match was fortuitous. Nevertheless, in this discussion, it is assumed that the model that fits the data the best is indeed the best model. This approach leads to the consistent application of professional judgment and avoids the "Black Box" approach of assuming that the Reference Man model is always the best model.
In vitro lung solubility and particle size information may greatly improve the accuracy of intake estimates. This information should be incorporated[xxxviii] into the evaluation if it is available. Modifications to parameters can be constrained by using several different types of radiobioassay data like feces, urine, and chest count data as it is important to get good agreement if multiple types of radiobioassay are used.

The major problem with iterative methods is that a biokinetic model, which includes exposure pathways and patterns, must be fully specified. Multiple over-lapping acute and chronic intakes can make it impossible to specify the model and perform an iterative evaluation. In many instances, non-iterative methods can be used to evaluate these cases.

[xxxviii] One of the advantages of using the ICRP 66 human respiratory tract model is the ease with which *in vitro* solubility data may be incorporated into the model.

6.8. Statistical Techniques Used in Intake Evaluations

Statistics are used in intake evaluations to help answer the following questions:

- Do I have the right biokinetic model?
- What is the best estimate of the intake (assuming I have the correct biokinetic model)?

6.8.1. The Right Biokinetic Model

The right biokinetic model may be a modification of a standard model that adequately fits all the available radiobioassay data. Statistical techniques may be used to aid the evaluator in deciding whether or not a fit is adequate (i. e. the right amount of "wrongness"), but it must be stressed that the evaluator is the final judge of adequacy.

6.8.2. The Best Estimate of Intake

The best estimate of an intake[61], I, in an unweighted least squares fit (ULSF) is assumed to be the one that minimizes the sum of the squares of the difference between the observed contents, o_i, of a radiobioassay compartment and the expected contents, $\hat{o}_i$. The sum of the squares (SS) is given by:

Equation 12, Sum of the Squares (SS)

$$SS = \sum (o_i - \hat{o}_i)^2$$

where o_i and $\hat{o}_i$ are understood to be a function of time. Three conditions must be met for the minimization of SS to give unbiased estimates of the parameters that have the smallest variance of all possible estimates:

- The data must be normally distributed
- There must not be any significant systematic errors, that is, the correct model must be used and there are no outliers in the radiobioassay data
- The variance of all the data must be the same.

If there are any major deviations from the first two conditions, the minimization of SS may produce biased estimates of the parameters and, therefore, should not be used. A major deviation in the third condition will produce unbiased estimates of the parameters, but they will not have the smallest variance of all possible estimates. In this case, if the relative variance of each datum can be estimated, the variance of the data may be stabilized by dividing each term of SS by the variance of the respective o_i to obtain a weighted least squares fit (WLSF). The reciprocal of the variance is referred to as the weighting factor w:

Equation 13, Weighted Least Squares Fit (WLSF)

$$SS = \sum w_i (o_i - \hat{o}_i)^2$$

For a given biokinetic model, ô is calculated using f, the fraction of a unit intake expected to be present at time, t, after an intake, I:

Equation 14, Fraction of a unit intake

$$\hat{\boldsymbol{o}}_{\boldsymbol{i}} = \boldsymbol{I} * \boldsymbol{f}_{\boldsymbol{i}}$$

One can substitute $I * f_i$ for $\hat{o}_i$ in the WLSF equation for SS, differentiate with respect to I, set the expression equal to zero, and solve for I to give the following expression for the WLSF estimate of I:

Equation 15, WLSF estimate of I

$$I = \frac{\sum_{i=1}^{N} w_i \cdot o_i \cdot f_i}{\sum_{i=1}^{N} w_i \cdot f_i^2}$$

The weighting factor is usually the inverse of the total variance of the radiobioassay measurement. The total variance is composed of the variance of the radiometric technique, the biological variance of the worker, and the variance between the biokinetic model and the worker. This overall variance is seldom known; so, various assumptions are made for the weighting factor in order to calculate the intake with the above equation. Typical assumptions include:

- The weighting factor is equal to the inverse square of the total propagated analytical uncertainty σ_t of each measurement (i.e., the biological variance is negligible)

$$w_i = 1/{\sigma_t}^2$$

- The weighting factor is a constant, k, for all measurements

$$w_i = k$$

- The weighting factor is inversely proportional to the measurement

$$1/w_i \propto o_i$$

$$1/w_i = k_. * o_i$$

- The weighting factor is inversely proportional to the expectation

$$1/w_i \propto \hat{o}_i$$

$$1/w_i = k_. * \hat{o}_i = k * I * f_i$$

The second assumption leads to an ULSF estimate of the intake (which is referred to as a uniform absolute error fit in IMBA)

Equation 16, ULSF estimate of intake

$$I = \frac{\sum_{i=1}^{N} o_i \cdot f_i}{\sum_{i=1}^{N} f_i^2}$$

The third assumption leads to a point WLSF estimate of the intake (which is referred to as a square root error fit in IMBA)

Equation 17, WLSF estimate of intake

$$I = \frac{\sum_{i=1}^{N} f_i}{\sum_{i=1}^{N} \frac{f_i^2}{o_i}}$$

The fourth assumption leads to a group WLSF estimate of the intake (which is not directly available in IMBA):

Equation 18, Group WLSF estimate of intake

$$I = \frac{\sum_{i=1}^{N} o_i}{\sum_{i=1}^{N} f_i}$$

It is important to note that the value of the proportionality constant, k, does not influence the value of the intake.

A group WLSF is not directly available in IMBA, but it may be implemented by fitting the data by one of the available methods, substituting the intake retention fractions for the errors, and then performing a point WLSF. IMBA also offers other fitting methods like maximum likelihood (which often gives the same intake estimate as least-squares fitting) and Bayesian inference.

Finally, there is the traditional method of fitting the data by eye, which is often whimsically referred to as an "eye-chi" fit. Although maligned by many as being “unscientific,” a fit to data by the expert eye may offer the **best** chance of getting reasonable results if there are significant

systematic errors in the radiobioassay data or the model. The eye-chi fit will often provide the best "looking" fit, but the user should be aware that changes in scale could greatly affect what looks good.

There is often animated discussion among internal dosimetrists as to which is the best type of fitting procedure. If the correct biokinetic model is used and the data are well behaved, the expected and observed values will match rather well. In this case, the calculated intake will not be greatly affected by the type of fit used. On the other hand, if an incorrect model is used or the data are ill behaved, then the intake will vary considerably depending on the fitting technique used. In other words, if the fitting technique makes a big difference, you will probably get the wrong answer regardless of the fitting technique used.

In summary:

- The goal is to select a standard model that matches all observed data (i.e., the residual plot is uniform and random)
- If necessary, modify the biokinetic model to achieve an acceptable fit
- Although there is no preferred fitting technique, the point WLSF is recommended as the default fitting method
- If several models seem to fit the data equally well, the best (as opposed to correct) model is assumed to be the one that is the least complex[xxxix]. For example, if all of the available data are explained equally well with one intake of material A or two intakes of a material B, then the single intake scenario is preferred.

6.9. Less-Than Data

The term "less-than data" refers to data that are reported as less than some reporting or decision level. For example, plutonium urine radiobioassay can be reported as < 0.1 dpm/L and a ^{144}Ce whole body count as <20 nCi.

Less-than data are typically used as a constraint on a fit; the predictions of a model should agree with the less-than data. For example, if a model predicts a urine concentration of 0.002 dpm/L and the measured concentration is < 0.1 dpm/L, the empirical and expectation results are in agreement. Less-than data are not used for residual plots, or least squares fitting procedures. Alternatively, IMBA[xl] provides a maximum-likelihood fit that allows less-than data to be incorporated analytically into an intake calculation.

[xxxix] Occam's Razor - the principle that entities should not be multiplied needlessly; the simplest of two competing theories is to be preferred

[xl] See *IMBA User Manual Appendix A: Section A.10.4 Application to <LOD Data* for a discussion of this technique.

Less-than data are typically not used at SNL, and all results are typically reported as numerical values.

6.10. Outliers

An outlier is a datum that does not seem to belong with the rest of the data (i.e., it is significantly higher or lower than its neighboring data). In general, data is not rejected or "thrown out" because it does not seem to belong. This is because:

- Radiobioassay data tend to have considerable scatter and the data in close agreement, and not the outlier itself, may be the statistical anomaly
- The rejection of data from a least squares fit is typically subjective and has no statistical basis
- Additional data collected in the future may support the rejected datum (and maybe a new model).

The preferred method of handling apparent outliers is to collect sufficient data to "dilute" the influence of the suspect result. Sometimes, sufficient data cannot be obtained to dilute the outlier. In these cases the influence of the outlier on the fit can be reduced by giving it a relatively small weight or by using an eye-chi fit. Alternatively, the datum may be excluded from the intake calculation, but it should never be expunged from the evaluation[xli]. The goal here is to keep the datum "on the plot" so that it is not forgotten and can be brought back into the fitting process in the future if possible.

An outlier can also be a datum that does not agree with a particular model that was chosen because it agrees well with all the other data. If good reasons can be given for outliers that do not match a model, the datum may be given little weight in the evaluation. For example, the excretion on the first few days following an intake typically does not agree with models because the short-term excretion has the most uncertainty. For this reason, we do not have to be overly concerned with early excretion data that fails to support an otherwise consistent model.

6.11. Evaluation of Multiple Types of Radiobioassay Data

Two or more types of radiobioassay data may be available for evaluating an intake. For example, urine radiobioassay and feces radiobioassay data may be available following an intake of plutonium. The biokinetic model used to evaluate the intake should produce expectation radiobioassay results that agree with all the available observed radiobioassay data. In the plutonium example, the biokinetic model should produce expectation urine, and feces results that agree with the observed radiobioassay data. In practice, precise agreement with all observed data is seldom achieved. In these cases, an attempt is made to modify the model so as to produce an

[xli] IMBA allows data to be selectively excluded from an intake calculation but included on the plot of observed and predicted radiobioassay.

acceptable fit to all the observed data. If it is not feasible to obtain good agreement with all the data, we force the fit to match the "primary" radiobioassay data. The primary radiobioassay data are the data that are most indicative of the intake and hence should be given greater weight in the evaluation. For example, following an intake of plutonium, feces data can be used to help select a particle size and solubility, but the models and intakes are selected to best fit the urine data. Of course, the fit to the secondary radiobioassay data may degrade if the fit to the primary data is forced.

6.12. Assumptions Used for Calculating Intakes

Many assumptions may be required for evaluating occupational intakes of radioactive material because the biokinetic models can be very complex and limited data is available. Assumptions used to determine initial dose estimates should be conservative, producing the highest intake consistent with known facts. Follow-up radiobioassay programs should be designed to supply information that will permit conservative assumptions to be replaced with experimentally determined facts. Assumptions used for final intake calculations should be reasonably conservative, considering known facts and previous experience.

When information is not available, the following conservative default assumptions should be used to evaluate intakes for the purpose of determining initial follow-up actions:

- Intake pathway – inhalation
- Intake pattern – acute
- Aerosol AMAD - 5.0 µm
- Time of intake - immediately after the time when the last radiobioassay measurement was below the detection level or when the potential for exposure to radioactive materials began
- Absorption type - the type that results in the highest CED

If the absorption type of an actinide cannot be inferred from the radiobioassay data, the default ICRP absorption type should be assumed.

6.12.1. Determining the Most Probable Date of Intake

To evaluate radiobioassay data, it is necessary to know the point in time when the radioactive material entered the body. The time of intake is quite obvious when special radiobioassay is performed in response to a known incident. On the other hand, the time of intake is anything but obvious for intakes detected by routine radiobioassay and often a time of intake must be assumed.

When an intake is detected by routine urine radiobioassay, the first routine urine sample above the decision level (DL) and the preceding urine sample below the DL typically bound the time of

intake[xlii]. The appropriate Radiation Protection Line Support Team Project Leader (RPLST PL) investigates the activities of the worker during the time range bounded by the urine samples and reports this information back to the Internal Dosimetrist. From this investigation they determine a most probable date of intake (MPD). The MPD, as determined by an investigation, is preferred over an MPD determined by any other method. The radiobioassay data itself can be used to help pinpoint the MPD. For example, the presence of a short-lived component in the urinary excretion curve can indicate the MPD to within a month or so, and the investigation can be narrowed down to that time span. Also, the ratios of material in the body and excreta can help identify the source term and the time and place where the intake occurred.

For small intakes, even the most detailed investigations seldom indicate an MPD. In these cases, the first preference is to select the MPD to produce the best fit to the observed radiobioassay data. Note that this method can indicate a date for the MPD when the worker was not working with radioactive materials or was not even at work. No attempt is made to match the MPD to a time when the worker was actually working with radioactive materials in this case because:

- Attempting to tie the intake to a particular date has already proved unsuccessful

- The intake and dose are not significantly impacted by changes in the time of intake on the order of several weeks.

Finally, when all else fails, the MPD is defined as the midpoint between the two urine samples that bound the intake date[62]. This assumes that on every day the worker had an equal probability of having the intake and will, on average, lead to the most accurate intake estimates[63] for actinides if the other methods fail. However, improvements in access control and RTWD sign-in data in the future could make it feasible to quickly compile detailed histories of work with radioactive materials. Access to this information greatly facilitates the Radiological Protection Department's investigation. Specifically, when RWP sign-in dates in the time span bounded by the two urine samples are known the MPD is simply the mean of the sign-in dates. Calculating the MPD in this manner places more weight on the dates when it is known that the worker was working with radioactive materials and is considered to give a better estimate of the MPD than the midpoint method. The midpoint method will continue to be used when RWP sign-in dates are not known. A MPD determined by the midpoint does not have to coincide with a date when work with radioactive materials was actually performed.

In summary, when routine urine radiobioassay results above the DL are not associated with known events, the most probable date of intake is selected by one of the following methods that are given in the order of preference:

- The Internal Dosimetrist selects the MPD (in the time range bounded by the two urine radiobioassay results) based on an investigation of work activities by Radiological Protection. This investigation should take advantage of RTWD sign-in dates where possible.

[xlii] The radiobioassay could be whole body counts, etc., but typically this scenario is encountered with routine urine radiobioassay.

It is important to note that only a MPD determined by investigation must coincide with an actual date when work with radioactive materials was performed.

- The internal dosimetrist selects the MPD (in the time range bounded by the two urine radiobioassay results)[xliii] to produce the best fit to observed radiobioassay data

- The MPD is assumed to be the mean of RTWD sign-in dates

- In the absence of RTWD sign-in dates, the MPD is assumed to be the midpoint of the two urine radiobioassay results.

6.13. Uncertainties and Precision in Intake Estimates

Calculating an intake is an intermediary step between radiobioassay analysis and dose assignment. In this sense, the number of significant digits with which the intake is reported is not of paramount importance; however, to promote consistency and to avoid the appearance of unjustified precision, all intakes are officially reported with no more than three significant digits.

All intake (and therefore dose) estimates have a degree of uncertainty caused by random and systematic errors in the biokinetic models, the radiobioassay measurements, etc. These uncertainties are discussed in NCRP Report 87[64] and Traub and Robinson[65]. Calculated uncertainties in intake estimates are used to select the best model but are typically not reported for the following reasons:

- A "true" uncertainty is difficult or impossible in practice to determine

- There is no regulatory role for the uncertainties (i.e., they are not propagated to dose)

The ICRP[66,67] has acknowledged these problems in that they have not established requirements for accuracy in internal dose assessment. Preliminary studies at the Savannah River Test Site[68] have shown that, for large intakes of plutonium, errors in CED on the order of 50% might be expected, but that committed organ dose equivalent may have errors an order of magnitude larger. In internal dosimetry intercomparisons[69,70,71] the coefficient of variation for the dose estimates from 11 dosimetrists was on the order of 15% to 80% for intakes of various radioactive materials. When four internal dosimetrists evaluated an intake of ^{238}Pu that occurred at SRTS[72], a coefficient of variation of 38% was reported.

[xliii] In rare occasions, the best fit to the radiobioassay data is produced assuming that the intake occurred before the time period bounded by the two urine samples. Such assumptions are left to the professional judgment of the internal dosimetrist and should be justified in the case narrative.

6.14. Time Required for Intake Evaluations

The length of time required to evaluate a significant intake[xliv] following an incident is of intense interest to the worker(s) involved, management, and regulators. To accurately evaluate a significant intake of an actinide typically requires the establishment of an excretion or retention curve over a period of months or even years[73]. However, preliminary intake and dose estimates are usually issued long before the evaluation is "completed." Experience has shown that the sooner after an incident an intake estimate is made, the more likely it is to eventually require a significant revision. For this reason, it is "…very important not to overreact to initial dose assessments, which may be revised either upward or downward when radiobioassay data over a period of weeks or months becomes available[74]." In practice, the timetable for issuing dose estimates is a function of the type and number of radiobioassay available and the precision of the dose estimate:

- Early estimates that an internal dose may exceed a given value, typically a regulatory limit, are based on a known incident and one radiobioassay result or a routine positive radiobioassay result over an action level (~5-10 days).

- An incident dose can usually be bracketed based on a known incident and the results from two special radiobioassay results or a routine positive radiobioassay result, a positive follow-up result, and a range of potential intake dates (~20-40 days).

- A preliminary dose[xlv] may be estimated from a known incident and three urine radiobioassay results or a routine positive, two follow-up results, and an official intake[xlvi] date (~30-60 days).

- The "final" dose estimate is made once the internal dosimetrist feels that the excretion and retention curves have been defined well enough to preclude major changes in dose estimate in the future. This may take many months for a significant intake.

Chelation therapy will greatly complicate the intake evaluation and may delay dose estimates, especially if a protracted therapy regimen is implemented.

6.15. Biokinetic Models and Software Programs

The ICRP 30 family of biokinetic and dosimetric models was used to evaluate radiobioassay data and calculate internal dose at SNL is estimated to have been between 1992 and 2000. The principal software programs which implemented these models were Mathcad and Mathematica. Both were previously used to calculate dose, and stopped being the primary method to calculate

[xliv] Thanks are extended to Robert Loesch, DOE EH-52 at the time, for valuable discussions concerning this topic.
[xlv] Initial dose estimates are typically slightly overestimated so that subsequent revisions are downward. Gross overestimates are as undesirable as gross underestimates.
[xlvi] The most probable intake date, which is provided by the Radiological Protection, may be very difficult to determine and delay the dose estimate.

intakes and doses when the software program IMBA[75] was officially adopted in 2007. IMBA, which evaluates bioassay data and calculates dose, is based on the ICRP 66/67/68/69 biokinetic and dosimetric models[xlvii]. A complete discussion of these models was initially provided in the user manuals for IMBA[76] and will not be reproduced here. It should be noted that effective January 2007, the Professional Plus version of IMBA was implemented and the initial USDOE version of IMBA was retired.

There are situations where IMBA is not used to calculate intakes and doses and other software programs are used. In particular, IMBA is not used to calculate doses from intakes of HTO, doses from PAS measurements, and doses from intakes of transuranics influenced by chelation therapy. In these situations other methods and software programs will be used and properly documented.

6.16. Radiation and Tissue Weighting Factors

10CFR835 requires the use of ICRP 60 based radiation and tissue weighting factors. These factors are incorporated into all dose conversion factors and software programs used to calculate internal dose. Table 26 and
Table 27 provide a comparison of current radiation tissue weighting factors from an historic perspective as well as presenting the most recent scientific guidance from the ICRP.

Table 26, Radiation weighting factors

Radiation	ICRP 30 (1977)	ICRP 60 (1990)	10CFR835 (2007)	ICRP 103 (2007)
Alpha	20	20	20	20
Proton	10	5	4	2
Beta	1	1	1	1
Gamma, X-ray	1	1	1	1

[xlvii] Note that the tissue weighting factors specified in 10CFR835.2 are used to calculate effective dose.

Table 27, Tissue weighting factors

Tissue or Organ	ICRP 30 (1977)	ICRP 60 (1990)	10CFR835 (2007)	ICRP 103 (2007)
Gonads	0.25	0.20	0.20	0.08
Breasts	0.15	0.05	0.05	0.12
Red Bone Marrow	0.12	0.12	0.12	0.12
Lungs	0.12	0.12	0.12	0.12
Thyroid	0.03	0.05	0.05	0.04
Bone Surfaces	0.03	0.01	0.01	0.01
Colon	N/A	0.12	0.12	0.12
Stomach	N/A	0.12	0.12	0.12
Bladder	N/A	0.05	0.05	0.04
Liver	N/A	0.05	0.05	0.04
Esophagus	N/A	0.05	0.05	0.04
Brain	N/A	N/A	N/A	0.01
Salivary Glands	N/A	N/A	N/A	0.01
Skin	N/A	0.01	0.01	0.01
Remainder	0.06 for each of 5 organs with the highest dose	0.05 applied to 10 specified organs and tissues	0.05 applied to 10 specified organs and tissues	0.12 applied to the mean dose of 13 organs and tissues

6.17. Intake-to-Dose Conversion Factors

The intake-to-dose conversion factors in
Table 28 reflect the current ICRP 60-based tissue weighting factors listed in 10CFR835 for the SNL Source Term.

Table 28, Intake-to-dose conversion factors (DCFs)

	Sv/Bq			mrem/nCi		
	S	M	F	S	M	F
^{241}Am		2.70E-05			9.99E+01	
^{241}Pu	8.45E-08	5.83E-07		3.13E-01	2.16E+00	
^{239}Pu	8.37E-06	3.26E-05		3.10E+01	1.21E+02	
^{238}Pu	1.06E-05	3.01E-05		3.92E+01	1.11E+02	
^{238}U	4.73E-06	1.65E-06	5.84E-07	2.12E+01	6.11E+00	2.16E+00
^{236}U	6.32E-06	1.9E-06	6.16E-07	2.34E+01	7.03E+00	2.28E+00
^{235}U	6.11E-06	1.82E-06	6.05E-07	2.26E+01	6.73E+00	2.24E+00
^{234}U	6.83E-06	2.11E-06	6.51E-07	2.53E+01	7.81E+00	2.41E+00
^{60}Co	1.62E-08	7.12E-09		5.99E-02	2.63E-02	
^{90}Sr	7.71E-08		3.02E-08	2.85E-01		1.12E-01
^{137}Cs			6.91E-09			2.56E-02
OBT						
Tritium gas						
Tritiated water						

Table 29, Intake to dose conversion factors (DCFs) for gases

	Sv/Bq	mrem/nCi
OBT	4.10E-11	
Tritium gas	1.80E-15	
Tritiated water	1.80E-11	

6.18. Work Restrictions

The Internal Dosimetrist may recommend work restrictions if:

- The dose to a worker following an incident is not known, i.e., it is still under evaluation
- An internal dose assignment causing the worker to exceed the ACL[77]
- The lifetime dose of the worker exceeds his age in rem[78]
- The worker is excreting or has retained sufficient quantities of radioactive material so as to interfere with the detection and assessment of future intakes of radioactive material

Recommended work restrictions are made in writing to the worker's manager who is responsible for imposing the restrictions. The work restriction will include a request for the appropriate radiobioassay to ensure that observed data continue to agree with predicted values.

6.19. Calculating Intake and Dose from Air Monitoring Data

The results of general air samplers (GAS) are typically used to detect inadvertent releases of radioactive material in the workplace and possibly trigger special radiobioassay programs. GAS measurements are usually inadequate to estimate intakes because the measurements are not representative. In this context, representative means that the concentration of aerosols in the air measured by the GAS is identical to the concentration of aerosols in the air breathed by the worker. GAS measurements may not be representative because the sampling head is not in a worker's breathing zone for the duration of the exposure. On the other hand, a PAS is considered to representative by definition[xlviii]. The PAS is representative because it is in the breathing zone of a worker and moves with the worker for the duration of the exposure.

For internal dosimetry applications, air monitoring data are preferred to be interpreted in terms of exposure (DAC-hr). A DAC-hr is the amount of radioactive material Reference Man would inhale if the worker were exposed to an air concentration of 1 DAC for 1 hour. For example, 1 DAC-hr of ^{239}Pu is:

Equation 19, DAC-hr

$$DAC\text{-}hr = (5x10^{-12}\frac{\mu Ci}{cc})(1000\frac{cc}{L})(20\frac{L}{\min})(60\min) = 6x10^{-6}\,\mu Ci\,,$$

or 6 pCi. Notice that DAC-hr has the units of activity. A worker, a CAM, and a PAS are all air sampling devices that draw volumes of air at different rates:

	Flow rate (L per min)
Worker	20
CAM	57
PAS	4

If each of these air sampling devices is present in a room that has an air concentration of 1 DAC for 1 hour, the worker will inhale 6 pCi, the CAM will "inhale" (57/20) x 6 pCi = 17.1 pCi, and the PAS will "inhale" (4/20) x 6 pCi = 1.2 pCi (or 1/5 of what the worker inhaled). This means that the activity present on the CAM filter or PAS filter is directly proportional to the worker's intake. Note that that there is no need to know the length of time the sampler has been running as long as it is assumed that both the air sampler and the worker were exposed to the same atmosphere for the same length of time. For example, if the activity on a PAS filter is 100 pCi, then the exposure is

[xlviii] Per NUREG 1400, via *DOE Radiation Protection Programs Guide*, DOE G 441.1-1B, 3/1/2007

Equation 20, Exposure calculated from PAS filter activity

$$\frac{\left(\frac{20\,lpm}{4\,lpm}\right)(100pCi)}{\left(6\,pCi/DAC-Hr\right)} = 83.3DAC-hr$$

Appropriate respiratory protection factors are applied to the exposure measured by PAS. PAS are worn outside of full-face respirators and inside of anti-contamination clothing and fresh air hoods. Therefore, an exposure measured with PAS worn with a full-face respirator is divided by the appropriate RPF[xlix] before it is assigned to the worker.

At SNL, a PAS is considered to be a device that measures internal dose, and activity above the decision level on a PAS filter is a positive indication of an intake. This intake will deliver a dose, of some magnitude, which depends on the quantity and type of radionuclide inhaled. Per Federal Rule 10CFR835.702(b) and 10CFR835.209(b), this dose is assigned to the individual unless radiobioassay data can provide a more accurate estimate of the dose. In practice, this means:

- If an intake is measured with a PAS, and radiobioassay data cannot refute the intake, the intake measured by the PAS and its resulting dose are assigned to the worker.

- If an intake is measured by PAS, the radiobioassay data are capable of refuting the intake, and the radiobioassay do indeed indicate that an intake did not occur, then no dose is assigned to the worker from the PAS measurement in question.

- If both the PAS and the radiobioassay indicate that an intake occurred, both are used by the internal dosimetrist to arrive at the best estimate of the intake.

In Figure 2, intakes of uranium measured with PAS are plotted as a function of time relative to the date of the first use of PAS by the worker. In Figure 3, the urinary excretion expected to be produced by these intakes (assuming Type S material) is presented as a function of time. Figure 3 clearly shows that unless the urine sample was taken immediately after an exposure, the intake could not be quantified by urine radiobioassay. The application of the MDA in this case is overly optimistic because it ignores the problem of differentiating an occupational exposure of uranium from an environmental exposure. Thus, for low level exposures typical with PAS, the routine radiobioassay program is incapable of refuting an intake.

[xlix] 50 for a full-face respirator.

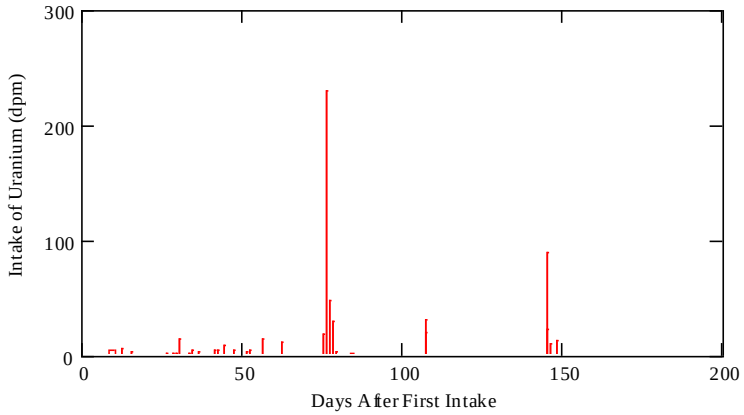

Figure 2, Intakes measured with PAS versus time

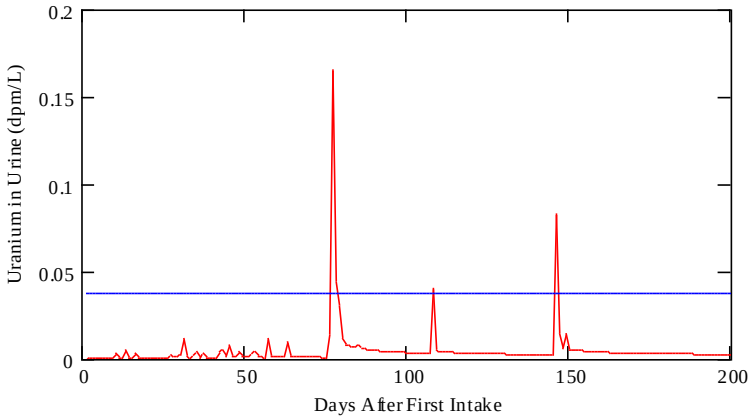

Figure 3, Urinary excretion expected to be produced by intakes. The blue line is an assumed detection level for ^{234}U

Special radiobioassay programs are typically triggered by exposures of more than 8 DAC-hr in any day or a cumulative exposure of more than 40 DAC-hr. Any exposure measured by PAS that is refuted by radiobioassay is not assigned to the worker and is therefore not included in the 40 DAC-hr action level. For example, assume a worker inadvertently touched the PAS filter while working in a radiologically controlled area, the PAS indicates an exposure of 10 DAC-hr of plutonium occurred, and a special radiobioassay program is promptly started. If the special radiobioassay program indicates that no exposure occurred, the 10 DAC-hr is not assigned to the worker and it does not count toward the 40 DAC-hr cumulative for the year.

Doses are calculated from PAS measurements by applying appropriate DCFs, like those in

Table 28, to the calculated intake. For example, 1 DAC-hr of ^{239}Pu measured by a PAS represents an intake of 0.006 nCi, which will deliver a CED[l] of (121 mrem/nCi) x (0.006 nCi) = 0.73 mrem. Note that 1 DAC-hr does not equal 2.5 mrem as one might expect because the ^{239}Pu

[l] Assuming 5 µm AMAD Type M ^{239}Pu

DAC is based on non-stochastic effects and the DCFs given above are based on the ICRP 60 family of

biokinetic and dosimetric models, not the ICRP 30 models. Using the DCF, rather than the ALI or DAC, to calculate dose eliminates the problems associated with round off errors in the ALIs and DACs.

At SNL, a reporting level of 10 mrem CED is employed for doses from intakes of non-tritium radioactive material. To be consistent with this reporting level, an annual reporting level of 10 mrem is used for PAS. This means that, if the dose received during the year is less than 10 mrem, it is truncated to 0 mrem. Doses above the annual reporting level are rounded to the nearest mrem for final reporting.

6.20. Significant Digits in Dose Estimates

Currently, dose is rounded to the nearest mrem before it is reported. Once rounded, the dose becomes an exact integer. A dose of 155,130 mrem, thus, has six significant digits. This is merely a bookkeeping practice and does not imply that doses can be determined with an error of one part in a million.

6.21. Cutoffs for Assigning and Reporting Dose

All identified intakes are evaluated as occupational dose. A reporting level of 10 mrem is used at SNL for internal dose from intakes of non-tritium radionuclides. This cutoff was established to simplify bookkeeping under an annual dose system by eliminating the tracking of doses that were insignificant with respect to SNL reporting levels and action levels. In practice, the reporting level means that if the CED from an intake of radioactive material is less than 10 mrem, then the person is informed in writing of this result and no dose is entered into the records. The basic premise of this practice is that intakes of non-tritium radionuclides are rare events and that no more than 10 mrem would go unreported in any given year.

The 10 mrem reporting level is applied to doses from intakes of tritium and from intakes measured by PAS (as discussed previously). These doses are not rare events. Therefore, to keep the unreported dose to less than 10 mrem per year, a 10-mrem reporting level is applied to the annual doses from intakes of tritium and intakes measured by PAS. Theoretically, a worker could potentially have up to 30 mrem go unreported in a year – 10 mrem from a non-tritium intake, 10 mrem from tritium intakes, and 10 mrem from intakes measured by PAS. However, such an event is so unlikely that it is assumed that the maximum unreported internal dose for any worker is 10 mrem per year[li].

[li] Note that this is unreported dose from detected intakes, not unreported dose from undetected intakes.

6.21.1. Organ and Tissue Dose Equivalent

Organ and tissue dose equivalents are calculated for all intakes as part of the effective dose calculation. However, up until 1999, organ and tissue dose equivalents were not reported to a worker unless the dose exceeded a deterministic limit or the individual requested the dose[79]. The

intent of this policy was to simplify the information that had to be communicated to the worker and enhance their ability to judge the significance of an internal dose. This policy was changed[80,81] in 1999 to require that organ and tissue doses be reported to workers when the internal dose is assigned.

6.22. Calculation of Dose to the Embryo/Fetus

In the unlikely event that it would be necessary to calculate dose equivalent to the embryo/fetus from a maternal intake of radioactive material, ICRP 88[82] would be utilized. An additional resource that may be consulted is NCRP Report 128[83].

6.23. Review of Intake and Dose Evaluations

All intake and dose calculations are performed by an internal dosimetrist using the methods described in this manual. All evaluations receive an internal peer review by a second internal dosimetrist or an external review by a second party. If the CED is equal to or greater than 100 mrem it will be presented to dosimetry management prior to final approval.

A dose assessment equal to or greater than 100 mrem CED is considered final only after it is approved by all parties involved in the review process. The conclusions of the review are documented with the dose and investigation results. For regulatory purposes, a dose is considered to be officially reported[lii] only after it has been communicated to the worker. Note that every effort will be made to inform the worker of the dose before it is reported to SNL and DOE management.

6.24. Historic Intakes and Dose Re-Evaluations

All intakes of radioactive material that occurred after January 1, 1989 were required by DOE Order 5480.11 to be evaluated in terms of annual effective dose equivalent (AEDE). Intakes that occurred before this date were "grandfathered," and did not have to be reevaluated in terms of AEDE. These intakes may be referred to as "historic" intakes. The DOE complex officially changed from an AEDE based system to a CEDE based system starting on January 1, 1994, the date the Federal Rule 10CFR835 became effective. Under the CEDE based system, historic intakes do not contribute dose in the current year. Thus, there is no technical reason for evaluating historic intakes.

[lii] This means that the Non-compliance tracking clock starts once the worker is informed of the assigned dose.

Internal dosimetrists frequently have reason to review personnel dosimetry records. For example, when a worker is involved in a radiological incident and an intake is assigned, the entire radiobioassay history of the worker may be reviewed. Since 1987, all radiobioassay results, including those collected prior to 1987, have been interpreted using the current models and methods at the time. This practice may result in a small intake being assigned from radiobioassay data that was not considered significant at the time it was collected. It has been found in industry

practice that small, unreported historic intakes frequently generate undue concern with some workers if they are reported years later[liii]. Therefore, historic intakes are not typically reassessed with current methods. However, special requests by workers for a reassessment will be honored.

As of approximately 2000, all new intakes were evaluated using the ICRP 60 family of biokinetic and dosimetric models. This raises the issue of if and when intakes evaluated with ICRP 30 models should be reevaluated using the new ICRP 60 family models. The central policy is that the new models will be used for the evaluation of any previous intake that is reevaluated for any reason and that there will be no systematic reevaluation of all intakes. There are two primary reasons for this policy, which is consistent with DOE guidance[84]. First, the reevaluation of all active cases was not required in order to comply with the regulations. Second, there are simply insufficient resources currently allocated to reevaluate the known cases of intakes of non-tritium radionuclides.

6.25. Historic Dose Evaluation Methods

Assessment of internal dose for workers in the nuclear weapons complex has always been based on the evaluation of radiobioassay data rather than air monitoring data. The methods used to evaluate radiobioassay data throughout the history SNL have changed to keep pace with improvements in the technology of internal dose assessment and the evolution of internal dose regulations. All radionuclides have not been impacted equally as regulatory changes have occurred. For example, the methods used to evaluate of intakes of tritiated water have not changed significantly in over 40 years whereas the methods used to evaluate intakes of plutonium have.

[liii] The question that is often asked is something along the lines of "…why didn't you tell me about this in 1963?"

7. SYSTEM PERFORMANCE AND TESTING

7.1. General Requirements

SNL shall take appropriate quality assurance measures to assure that internal dosimetry monitoring programs are adequate to meet the requirements of 10 CFR 835. Guidelines for appropriate quality assurance for internal dosimetry programs are contained in Section 11 of DOE Technical Standard DOE STD-1121-98 "Internal Dosimetry." RPDP ensures its radioanalytical service providers maintain quality programs consistent Department of Energy Laboratory Accreditation Program (DOELAP) requirements. It is RPDP's intent to meet the standards of, and apply for accreditation for internal dosimetry programs when such accreditation becomes available in the DOE community.

7.1.1. DOELAP Accreditation

Although no individuals are deemed likely to exceed 100 mrem CED from SNL work activities, RPDP maintains DOELAP accreditation for indirect and direct radiobioassay analyses through the Department of Energy Laboratory Accreditation Program (DOELAP). The accreditation involves submittal of radiobioassay lab documentation to DOELAP by RPDP, triennial performance testing of each of the two analytical labs that are tied to RPDP's DOELAP accreditation, and an onsite assessment of each lab by DOELAP technical assessors. A summary of the DOELAP accreditation categories and analytes for which RPDP is accredited appears in
Table 30 and Table 31. RPDP's DOELAP radiobioassay accreditation letter, certificate, and conditions of accreditation for direct and indirect bioassay services is posted in the RPDP Lab in Building 869/B13. A copy is maintained on the IH & RP Labs website (on the Sandia Restricted Network, SRN).

Table 30, SNL Radiobioassay Indirect DOELAP Accreditation Nuclides

		Urine (TAL)	Feces (TAL)	Urine (RPSD)	Feces (RPSD)
I	Beta Activity: Avg. Energy < 100 keV, H-3	--	--	Yes	--
II	Beta Activity: Avg. Energy < 100 keV, Sr-90	Yes	Yes	--	--
III	Alpha Activity, isotopic analysis				
	Thorium-228/232	Yes	Yes	Yes	--
	Thorium-230	Yes	Yes	Yes	--
	Uranium-234/235	Yes	Yes	Yes	--
	Uranium-238	Yes	Yes	Yes	--
	Neptunium-237	Yes	Yes	Yes	--
	Plutonium-238	Yes	Yes	Yes	--
	Plutonium-239/240	Yes	Yes	Yes	--
	Americium-241	Yes	Yes	Yes	--
IV	Elemental mass/volume-Uranium	--	--	Yes	--
V	Gamma (photon) activity				
	Cobalt-60	--	--	Yes	--
	Iodine-125	--	--	Yes	--
	Cesium-137	--	--	Yes	--

Table 31, SNL Radiobioassay Direct DOELAP Accreditation Nuclides

		Type	Thyroid (RPSD)	Thyroid I-125 (RPSD)	WBC (RPSD)
IV	Fission and activation products				
	Cesium-134	Total body	--	--	Yes
	Cesium-137	Total body	--	--	Yes
V	Gamma (photon) activity				
	Iodine-125	Thyroid	--	Yes	--
	Iodine-131	Thyroid	--	Yes	--

7.2. Analytical Service Laboratories

The onsite RPSD laboratory and Sandia's offsite contracted radiobioassay services laboratory perform all direct and indirect analysis of bioassay samples and human subjects. Both are required by contract to maintain rigorous, extensive, well-documented QA and QC programs consistent with DOELAP requirements in DOE Standard DOE-STD-1112-98 (DOE 1998), including those incorporated by reference to ANSI N13.30(1996). Each lab is required to maintain a QA manual that outlines responsibilities and provides requirements for data control, document control, calibration and checks of maintenance and test equipment, procedures, training, corrective action in the event of noncompliance, and traceability to standardizing bodies such as the National Institute of Standards and Technology (NIST).

The QC program involves analyzing blanks with each batch of samples, urine spikes monthly, and fecal spikes quarterly, quality reviews of all data to ensure QC limits are met, and publishing annual QC reports. The blank and spiked samples are used to ensure that laboratories demonstrate continued proficiency to evaluate samples at MDA-level quantities for each routine analysis and matrix. Laboratories are required to demonstrate that actual MDAs are no greater than the levels specified in the contract and that bias and precision are within specified limits.

An annual audit of each lab is performed by RPDP and IH & RP Labs QA staff. The audit is performed in accordance with DOE-STD-1112-98 and ensures the lab is conducting analyses in accordance with the latest approved contract Statement of Work (SOW) for the SOW's routinely used analyses/matrices. All routine analyses (i.e., not research and procedure development work) must be done according to written and approved procedures. In addition, all analysts must be trained and certified in each procedure before they can routinely perform the applicable analysis.

7.3. Dose Assessments

7.3.1. Procedure Documentation

Methods and models used to assess intakes and doses shall be documented in the SNL Technical Basis Manual for Internal Dosimetry and as needed on a case-by-case basis. Program processes related to the implementation of this manual are documented in internal dosimetry program procedures and instructional job-aid documents.

7.3.2. Internal Dosimetrist Qualifications

Personnel with the technical responsibility for internal dose evaluation shall have the necessary expertise and experience based on appropriate education and training in conjunction with practical experience to perform their assigned duties. In general, the minimum requirement is a master's degree in Health Physics or closely-related field. Formal additional professional-level education and/or experience in internal dosimetry are highly desirable.

7.3.3. Review of Internal Dose Assessments

The evaluation and assessment of internal doses can be complex, and often involves a good deal of individual professional judgment. As noted in DOE-STD-1121-98, "Agreement within a factor of two among experienced dose assessors is probably the best that can be hoped for in difficult cases such as transuranic intakes with subsequent chelation."

All formal (non-automated) assessments of internal dose shall be technically reviewed by a second internal dosimetrist (or designated qualified Health Physicist) prior to submission for recording in a worker's dose of record. When necessary or desirable, an independent review of dose assessments will be obtained. An independent review such as this may be conducted by SNL or external personnel, but should be conducted by a person with recognized internal dosimetry expertise. RPDP maintains a contract with one or more internal dosimetry consultants for this purpose.

7.4. Internal Dosimetry Program Records

The records generated by the RPDP are maintained in files within the IH & RPS Labs organization. Archiving occurs periodically in conjunction with the SNL Customer Funded Records Center. Additional information on IH & RP Labs records requirements is provided in the *Records Management Plan*, DLPS-100.

7.5. Software Programs

7.5.1. General Considerations

A variety of computer software and calculation tools are used at SNL for internal dose evaluation. These tools may include programs developed and maintained "in-house", as well as custom-developed tools implemented using commercially available software such as MS Excel or Mathcad. These applications may be used for both intake/dose evaluation and data management.

7.5.2. Configuration Management

Many software programs used by dosimetry are subject to configuration management recording requirements. Configuration management records include:

- an identifying version number and applicable date
- a copy of the code or worksheet
- instructions for running the code or worksheet
- acceptance and/or validation records

7.5.3. Verification and Validation

Software programs used for internal dose evaluation should undergo verification and validation as necessary. Verification may involve determining the program requirements, the range of program results that may be considered valid, or criteria to be used in evaluating the validity of the results. Validation is the process of testing the program or worksheet under a specific computing system and evaluating the results to ensure compliance with the specified requirements. The core of this testing process is running of "benchmark" cases for comparison against an independently performed calculation (e.g., published results, hand calculations, etc.). Specific guidelines on verification and validation are presented in Section 11 of DOE-STD-1121-98.

7.5.4. Software Security

Backup copies of internal dosimetry software and data should be kept in a secure location. A second copy should be kept in a different location for purposes of disaster recovery.

7.6. Program Audits

Internal audits of all functional elements of the radiation protection program are required to be conducted at least every third year[85]. Audits of the internal dosimetry program are included in the SNL Radiation Protection Program triennial self-assessments. Self-assessments may include reviews of:

- program documentation
- program implementation
- dose assessment procedures
- data management
- recording and reporting
- qualifications of personnel
- adequacy of staffing and resources
- other key elements of an internal dose monitoring program, as necessary

These assessments are performed to assure that the program maintains the capability to provide quality radiation protection dosimetry measurements to SNL workers.

RPDP performs periodic informal reviews of the source term and facility/operation-specific individual monitoring programs to assure that the internal dosimetry program design rationale and operating procedures are appropriate to support the SNL mission.

7.7. Internal Dosimetry Program Performance Metrics

Internal Dosimetry will track and report programmatic leading indicators related to the implementation of and compliance with this manual and the overall Internal Dosimetry Program in accordance with guidance from Radiation Safety Section and Hazards Control management. Examples of internal dosimetry programmatic leading indicators may include:

- Positive/anomalous result notification time
- Number of internal dosimetry result investigations

- Bioassay result follow-up and closure times
- Dose assessment completion time
- Worker sampling compliance

Use of tracking such leading indicators is intended to address any chronic issues potentially impacting the internal dosimetry program health and general compliance with the internal monitoring program. Any acute issues that would significantly impact RPDP's ability to perform prompt result notifications and timely assessments are reported to the IH & RP Labs Manager.

GLOSSARY

absorption type	A physical property of a material that relates to the dissolution rate and translocation from the respiratory tract to blood. ICRP 66 lists three absorption types: Type F (fast rate of absorption), Type M (moderate rate of absorption), and Type S (slow rate of absorption).
APF	Assigned protection factor. The expected level of protection that would be provided by a properly functioning respiratory protection device to properly fitted and trained users. Operationally, the assumed inhaled concentration can be estimated by dividing the ambient airborne concentration by the APF.
ALI	Annual limit on intake. The derived limit for the amount of a radionuclide taken into the body or a worker in a calendar year. The ALI is the smaller of the values that would result in a CED of 5 rem or a CEqD of 50 rem to any single organ or tissue.
AMAD	Activity Median Aerodynamic Diameter. A particle size where fifty percent of the total activity of the aerosol is associated with larger particle sizes.
burden	The quantity of radionuclide(s) in an organ or tissue at a stated time. Often referred to as the retained quantity.
CEqD HT(50)	Committed equivalent dose. The calculated dose to an organ or tissue for the 50 year period post intake.
CED E(50)	Committed effective dose. The effective dose for the 50 year period post intake.
compartment	A mathematical representation of an organ or tissue through which radioactive material can be deposited, retained, excreted, or transferred. Blood is commonly referred to as the transfer compartment.
DAC	Derived air concentration. The airborne concentration that equals the annual limit on intake for a radionuclide divided by a worker breathing for a working year (nominally taken to be 20 L*min-1 x 2000 hr = 2400m3)
DAC-hr	The equivalent exposure to one derived air concentration (DAC) of a radionuclide for a period of one hour. Note that it is not a step function but an integrated time and concentration value with units of activity.

decision level	The quantity of material in a measurement above which the analyte is declared to be detected. A decision level is a statistic of a single measurement.
decorporation	The removal of radioactive material from the body via physical and/or chemical means.
deposition	The material deposited at an entry site at a stated time.
deterministic	An effect directly related to the severity of a radiation exposure.
effective dose	The summation of the products of equivalent doses multiplied by the appropriate tissue weighting factors.
injection	Any route where radioactive material is put in direct contact with blood, excluding the lung or gastrointestinal tract.
intake	The amount of a radionuclide that enters the body.
internal dose	Dose received from radioactive material taken into the body.
in vivo	From Latin meaning "in life. " The direct measurement of radioactivity in a worker. An example would be a chest count.
in vitro	From Latin mean "in glass. " The measurement of radioactivity from a sample collected from a worker. An example would be a urine sample.
MDA	Minimum detectable activity. The smallest amount of a radionuclide that will be detected with a certain degree of confidence (for example, 2σMDA is a statistic of the ability of an analytical method.
PAS	Personal air sample. A form of personal air monitoring that involves the sampling of air in the immediate vicinity (typically within one foot) of an individual's nose and mouth, usually by a portable sampling pump and collection tube (e.g. lapel sampler) worn on the body. PAS may be used to estimate exposure to personnel in accordance with 10CFR835.209 but must be representative (directly related to) of the air breathed by the worker.
radiobioassay	The determination of kinds, quantities, or concentrations, and, in some cases, locations of radioactive material in the human body, whether by direct measurement or by analysis of radioactive materials excreted or removed from the human body.
retention	The retained quantity of material at a stated time. Often expressed as a fraction of the intake or uptake.

stochastic	A malignant or genetic effect for which there is a probability of the effect having resulted from radiation exposure.
TWF	Tissue weighting factor The fraction of the overall health risk, resulting from uniform, whole body irradiation, attributable to a specific tissue.
Total effective dose	The sum of effective dose from external exposures and internal exposures (CED).
transportable	Refers the to the relative rate of transfer from an initial deposition. Not always the same as chemical solubility.
uptake	Quantity of a radionuclide taken up by the blood or a specified organ or tissue via blood. Intake and uptake are often used interchangeably but are not the same.

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