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Dose Reconstruction  
Project for NIOSH**

Oak Ridge Associated Universities | NV5|Dade Moeller | MJW Technical Services

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**Savannah River Site – Occupational Internal  
Dose**

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Subject Expert(s): Paul J. Demopoulos, Roger L. Halsey, and Sallie D. Robinson

Document Owner Approval:	<u>Zachariah C. Tribbett Signature on File for</u> James M. Mahathy, Document Owner	Approval Date:	<u>03/01/2024</u>
Concurrence:	<u>Signature on File</u> John M. Byrne, Objective 1 Manager	Concurrence Date:	<u>03/01/2024</u>
Concurrence:	<u>Signature on File</u> Scott R. Siebert, Objective 3 Manager	Concurrence Date:	<u>03/01/2024</u>
Concurrence:	<u>Vickie S. Short Signature on File for</u> Kate Kimpan, Project Director	Concurrence Date:	<u>03/01/2024</u>
Approval:	<u>Signature on File</u> Timothy D. Taulbee, Associate Director for Science	Approval Date:	<u>03/05/2024</u>

**FOR DOCUMENTS MARKED AS A TOTAL REWRITE OR REVISION,  
REPLACE THE PRIOR REVISION AND DISCARD / DESTROY ALL COPIES OF THE PRIOR REVISION.**

New                       Total Rewrite                       Revision

**PUBLICATION RECORD**

<b>EFFECTIVE DATE</b>	<b>REVISION NUMBER</b>	<b>DESCRIPTION</b>
07/15/2003	00	Original issue technical basis document for the Savannah River Site incorporates changes based on information gathered through contacts with SRS technical subject matter experts and incorporates NIOSH comments provided 7/11/2003. First approved issue. Initiated by Edward D. Scalsky.
08/21/2003	01	Revised to incorporate changes to Sections 2.2, 3.2.2, Table A-23, Tables B-1, C-17, and C-19; to add to the in vitro fission product analyses in Sections 4.1.2 and 5.3.5; to add dose reconstructor instructions in Sections 2.5, 2.5.1, 2.5.2; 3.4, 3.4.1, 3.4.2; 4.5, 4.5.1, 4.5.2, 4.5.3, 4.5.4; 5.7, 5.7.1, and 5.7.2 and to Tables D-11 through D-29. Additions made to Sections 4.4.3 and 4.4.4. Added Sections 4.4.3 and 4.4.4 to previous record of revisions. Corrected several terms and definitions in the Glossary and Sections 1.3 and 2.2. Revised Section 2.2 for clarity. Reassigned the thyroid to Group 2 in Section 2.5.2. In Section 4.1.2 revised MDA/reporting level in Tables 4.1.1-1 and 4.1.1-2, clarified 10-year-old 12 wt% Pu mix as favorable to claimants, added text to the Note about fission product <sup>90</sup> Sr sample and whole-body counts, and corrected absorption types. Added two paragraphs in Section 2.4.1 discussing record forms for whole-body counts and meaning of terms therein. Corrected Tables 4.4.3-1 and 4.4.3-3 type S and M values respectively and corrected units. Added bullets under note in Section 4.4.3, clarified tritium equilibrium and corrected calculation, and corrected solubility type in Tables 4.4.3-7 and 4.4.3-8. Added text to Sections 4.5 and 4.5.1 and deleted sections related to best-estimate approaches. Deleted footnote in Section 5.3.2.1. Added words to clarify use of annual intakes for dose reconstruction. Corrected Table D-5 absorption type. Initiated by Edward D. Scalsky.
10/29/2004	02	Revised to simplify internal dose reconstruction. Allows use of Table C-17 for all environmental dose reconstruction and clarifies periods of applicability. Modified instructions for determination of undetected dose from in vitro bioassay results (except tritium) below detection threshold. Clarified instructions for missed dose determination from whole-body count results below the detection threshold. Corrected <sup>144</sup> Ce Table 4.3-5. Pages changed are 24, 57, 58, 62, 72, 80, 81, 84, 85, 86, 89, 156 and 183. Deleted Section 4.4.2 on uncertainty because it is not specific to SRS and because this avoids inconsistencies that might result from dose reconstruction programmatic changes. Incorporated medical X-ray changes and addition of section on photofluorography and corrected period of applicability. Added references and information on TNX and CMX. Deleted paragraph related to consumption of venison.
04/05/2005	03	Revised to add Section 3.1.3 and Attachment C.3 to include ingestion of foodstuffs.
03/05/2024	04	Revised to update entire site profile and to separate it into six technical basis documents. This Part 5 was Section 4.0 of Revision 03. Revised to incorporate comments and guidance associated with issuance of two SEC classes implemented with Petition SEC-0103.

EFFECTIVE DATE	REVISION NUMBER	DESCRIPTION
		<p>Incorporates information from the internal co-exposure study (formerly ORAUT-OTIB-0081 and now ORAUT-TKBS-0003-7). Attributions and Annotations removed. Revised to include a thorough description and delineation of the types of uranium used and processed at SRS. Updated Section 5.6.4.2 to include reference to ORAUT-OTIB-0066 in OBT discussion and to edit unmonitored periods where co-exposure tritium dose is assigned in accordance with Table 7-18. Updated Section 5.6.3 to clarify whether and how to make comparisons between weapons-grade plutonium vs. nonmonotonic “pure” <math>^{238}\text{Pu}</math> for given facilities. Updated Section 5.6.5 to clarify where and how to apply the various uranium mixtures indicated tables other than NU and DU. Added a table in Section 5.6.5 that states which uranium mixture is to be applied for various facility codes by year range. Updated Section 5.6.9 to include wording to assume triple-separated thorium from 10/01/1972 to present. Updated Section 5.6.5.1 to clarify the “intakes to add” values in Table 5-31. Deleted “Uranium co-exposure intakes should be assigned to unmonitored workers while working in D Area from 1954 through 1988. Consult ORAUT-TKBS-0003-2 for rational.” from Section 5.6.5.3. Updated Section 5.6.7.1 to change 1988 to 1990 for the following statement: “FP co-exposure intakes should be assigned to unmonitored workers for work in D-Area from 1954 through 1988.” Incorporates formal internal and NIOSH review comments. Constitutes a total rewrite of the document. Training required: As determined by the Objective Manager. Initiated by John M. Byrne and authored by James M. Mahathy.</p>

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## ACRONYMS AND ABBREVIATIONS

ABRWH	Advisory Board on Radiation and Worker Health
AEC	U.S. Atomic Energy Commission
ANSI	American National Standards Institute
AWE	Atomic Weapons Employer
CEDE	committed effective dose equivalent
C.F.R.	<i>Code of Federal Regulations</i>
Ci	curie
cm	centimeter
CMX	Corrosion Mock-up Experiment
cpd	counts per disintegration
cpm	counts per minute
CTW	construction trade workers
CWT	chest wall thickness
d	day
DNC	delayed neutron counting
DOE	U.S. Department of Energy
DOL	U.S. Department of Labor
dpm	disintegrations per minute
DU	depleted uranium
EEOICPA	Energy Employees Occupational Illness Compensation Program Act of 2000
EU	enriched uranium
F	fast (absorption type)
FMPC	Feed Materials Production Center
FP	fission product
ft	foot
FWHM	full-width half maximum
g	gram
HP	health physics
HPRED	Health Protection Radiation Exposure Database
HPS	Health Physics Society
HTO	tritiated water
IA	induced activity or inconclusive analysis depending on context
ICRP	International Commission on Radiological Protection
IMBA	Integrated Modules for Bioassay Analysis
in.	inch
keV	kiloelectron-volt, 1,000 electron-volts
kg	kilogram
L	liter
m	meter
M	moderate (absorption type)
MCS	motor-controlled scanning

MDA	minimum detectable activity
mg	milligram
min	minute
mL	milliliter
mrem	millirem
MWd	megawatt-day
Nal(Tl)	thallium-drifted sodium iodide
nCi	nanocurie
ng	nanogram
NIOSH	National Institute for Occupational Safety and Health
NM	nonmonotonic
NNSA	National Nuclear Security Administration
NTA	nuclear track emulsion, Type A
NU	natural uranium
OBT	organically bound tritium
ORAU	Oak Ridge Associated Universities
ORAUT	ORAU Team
PAS	personal air sampling
pCi	picocurie
ppm	parts per million
PUREX	plutonium-uranium extraction
QC	quality control
RCA	Radiologically Controlled Area
RWP	Radiation Work Permit
S	slow (absorption type)
SD	standard deviation
SEC	Special Exposure Cohort
SMT	stable metal tritides
SRDB Ref ID	Site Research Database Reference Identification (number)
SRL	Savannah River Laboratory
SRNL	Savannah River National Laboratory
SRNS	Savannah River Nuclear Solutions
SRS	Savannah River Site
SS	super S (absorption type)
TBD	technical basis document
TEVA	trioctyl and tridecyl methyl ammonium-based resin
TIOA	tri-isooctylamine
TNX	Semi-works Laboratory
TRM	target residual material
TRU	transuranic
tU	ton of uranium
U.S.C.	<i>United States Code</i>
WB	whole body
WSRC	Westinghouse Savannah River Company



wt%	weight percent
yr	year
μCi	microcurie
μg	microgram
§	section or sections

## 5.1 INTRODUCTION

Technical basis documents (TBDs) and site profile documents are not official determinations made by the National Institute for Occupational Safety and Health (NIOSH) but are rather general working documents that provide historical background information and guidance to assist in the preparation of dose reconstructions at particular U.S. Department of Energy (DOE) or Atomic Weapons Employer (AWE) facilities or categories of DOE or AWE facilities. They will be revised in the event additional relevant information is obtained about the affected DOE or AWE facility(ies), such as changing scientific understanding of operations, processes, or procedures involving radioactive materials. These documents may be used to assist NIOSH staff in the evaluation of Special Exposure Cohort (SEC) petitions and the completion of individual dose reconstructions under Part B of the Energy Employees Occupational Illness Compensation Program Act of 2000 (EEOICPA).

In this document the word “facility” is used to refer to an area, building, or group of buildings that served a specific purpose at a DOE or AWE facility. It does not mean nor should it be equated to an “AWE facility” or a “DOE facility.” The term “AWE facility” is defined in EEOICPA to mean “a facility, owned by an atomic weapons employer, that is or was used to process or produce, for use by the United States, material that emitted radiation and was used in the production of an atomic weapon, excluding uranium mining or milling.” 42 *United States Code* (U.S.C.) § 7384I(5). On the other hand, a DOE facility is defined as “any building, structure, or premise, including the grounds upon which such building, structure, or premise is located—(A) in which operations are, or have been, conducted by, or on behalf of, the [DOE] (except for buildings, structures, premises, grounds, or operations ... pertaining to the Naval Nuclear Propulsion Program); and (B) with regard to which the [DOE] has or had—(i) a proprietary interest; or (ii) entered into a contract with an entity to provide management and operation, management and integration, environmental remediation services, construction, or maintenance services.” 42 U.S.C. § 7384I(12). The DOE determines whether a site meets the statutory definition of an AWE facility and the U.S. Department of Labor (DOL) determines if a site is a DOE facility and, if it is, designates it as such.

Under EEOICPA, a Part B cancer claim for benefits must be based on an energy employee’s eligible employment and occupational radiation exposure at a DOE or AWE facility during the facility’s designated time period and location (i.e., a “covered employee with cancer”). After DOL determines that a claim meets the eligibility requirements under Part B of EEOICPA, DOL transmits the claim to NIOSH for a dose reconstruction. EEOICPA provides, among other things, guidance on eligible employment and the types of radiation exposure to be included in an individual dose reconstruction. Under EEOICPA, eligible employment at a DOE facility includes individuals who are or were employed by DOE and its predecessor agencies, as well as their contractors and subcontractors at the facility. 42 U.S.C. § 7384I(11). Also under EEOICPA, the types of exposure to be included in dose reconstructions for DOE employees are those radiation exposures incurred in the performance of duty. As such, NIOSH includes all radiation exposures received as a condition of employment at DOE facilities in its dose reconstructions for covered employees, which may include radiation exposures related to the Naval Nuclear Propulsion Program at DOE facilities, if applicable. This is because NIOSH does not determine the fraction of total measured radiation exposure at a DOE facility that is contributed by the Naval Nuclear Propulsion Program at the DOE facility during a specified period of time for inclusion in dose reconstruction.

NIOSH does not consider the following types of exposure as those incurred in the performance of duty as a condition of employment at a DOE facility. Therefore these exposures are not included in dose reconstructions for covered employees [NIOSH 2010]:

- Background radiation, including radiation from naturally occurring radon present in conventional structures, and
- Radiation from X-rays received in the diagnosis of injuries or illnesses or for therapeutic reasons.

### 5.1.1 **Purpose**

This TBD discusses the occupational internal dose monitored workers might have received at the Savannah River Site (SRS).

### 5.1.2 **Scope**

The radionuclides identified by SRS as being the most significant in delivering doses or that were effective tracers for significant dose-producing radionuclides are presented. The methods for analysis of selected individual radionuclides are discussed. Tables providing the minimum detectable activities (MDAs) are included for the in vitro analysis of urine for selected radionuclides and in vivo whole body (WB) and chest counting.

Validation, verification, and support data for the figures and tables in this TBD are contained in Oak Ridge Associated Universities (ORAU) Team (ORAUT) supporting files [ORAUT 2023].

### 5.1.3 **Special Exposure Cohort**

#### **January 1, 1953, through September 30, 1972**

The Secretary of the U.S. Department of Health and Human Services has added the following class of SRS employees to the SEC [Sebelius 2012, p. 3]:

*All employees of the Department of Energy, its predecessor agencies, and their contractors and subcontractors who worked at the Savannah River Site from January 1, 1953, through September 30, 1972, for a number of work days aggregating at least 250 work days, occurring either solely under this employment or in combination with work days within the parameters established for one or more other classes of employees included in the Special Exposure Cohort.*

The Secretary based this designation on the findings of NIOSH's SEC Evaluation Report, which found it is not feasible to estimate internal exposures with sufficient accuracy for all externally monitored employees from January 1, 1953, through December 31, 1957, whose records have dosimetry codes A, G, CMX, or TNX. Further, NIOSH found that it lacked sufficient internal thorium monitoring data or other data or methods to support bounding internal thorium doses for SRS workers who may have worked with thorium in the 700-Area or the Corrosion Mock-up Experiment (CMX)/Semi-works Laboratory (TNX) Area from January 1, 1958, through September 30, 1972, whose records have dosimetry codes 5A, 5C, 6B through 6Z, 12D through 12H, or 12J through 12Z [NIOSH 2011].

#### **October 1, 1972, through December 31, 1990**

The Secretary has also added the following class [Becerra 2021, p. 3]:

*All construction trade employees of Department of Energy subcontractors [excluding employees of the following prime contractors who worked at the Savannah River Site in Aiken, South Carolina, during the specified time periods: E. I. du Pont de Nemours and Company, October 1, 1972, through March 31, 1989; and Westinghouse Savannah River Company, April 1, 1989, through December 31, 1990], who worked at the Savannah River Site from October 1, 1972, through December 31, 1990, for a number of work days aggregating at least 250 work days, occurring either solely under this employment or in combination with work days within the parameters established for one or more other classes of employees included in the Special Exposure Cohort.*

The Secretary based this designation on the recommendation of the Advisory Board on Radiation and Worker Health (ABRWH), which found that dose reconstruction for unmonitored subcontractor

construction trades workers (CTWs) who should have been monitored via the permit-driven job-specific monitoring program is not feasible using the co-exposure models NIOSH developed due to the nature of radiological work assigned to transient subcontractor CTWs, the lack of assurance provided their bioassay monitoring, and identified gaps in the permit-driven job-specific monitoring program. The ABRWH concluded that the completeness and representation of subcontractors who were, or should have been, monitored has not been sufficiently established [Anderson 2021].

Although the ABRWH found that it is not possible to completely reconstruct radiation doses for either class, NIOSH intends to use any internal and external monitoring data that might become available for an individual claim and that can be interpreted using existing NIOSH dose reconstruction processes or procedures to conduct partial dose reconstructions for employees who do not qualify for inclusion in the SEC [Anderson 2021; NIOSH 2011].

## 5.2 RADIONUCLIDES OF CONCERN

Radionuclides of concern in the *Savannah River Site Internal Dosimetry Technical Basis Manual* [Westinghouse Savannah River Company (WSRC) 2001], Chapter 5, include:

- Tritium ( $^3\text{H}$ ),
- Special tritium compounds,
- Manganese ( $^{54}\text{Mn}$ ),
- Cobalt ( $^{60}\text{Co}$ ),
- Zinc ( $^{65}\text{Zn}$ ),
- Strontium ( $^{90}\text{Sr}$ ),
- Ruthenium ( $^{106}\text{Ru}$ ),
- Antimony ( $^{125}\text{Sb}$ ),
- Cesium ( $^{134}\text{Cs}$ ,  $^{137}\text{Cs}$ ),
- Cerium ( $^{144}\text{Ce}$ ),
- Europium ( $^{152}\text{Eu}$ ,  $^{154}\text{Eu}$ ),
- Thorium ( $^{228}\text{Th}$ ,  $^{232}\text{Th}$ ),
- Uranium ( $^{234}\text{U}$ ,  $^{235}\text{U}$ ,  $^{236}\text{U}$ ,  $^{238}\text{U}$ , and mixtures),
- Neptunium ( $^{237}\text{Np}$ ),
- Plutonium ( $^{238}\text{Pu}$ ,  $^{239}\text{Pu}/^{240}\text{Pu}$ ,  $^{241}\text{Pu}$ ),
- Americium ( $^{241}\text{Am}$ ,  $^{243}\text{Am}$ ),
- Curium ( $^{244}\text{Cm}$ ), and
- Californium ( $^{252}\text{Cf}$ , including  $^{248}\text{Cm}$  progeny).

SRS identified these radionuclides as the most significant at delivering doses or that were effective tracers for significant dose-delivering radionuclides. However, SRS produced other radionuclides (sometimes referred to as exotics) in the C and K reactors. These radionuclides were generated by the irradiation of samples of various materials used for research purposes at various universities and laboratories. SRS produced over 150 different radionuclides of 66 elements from 1965 to 1970. Samples were in welded-closed aluminum cans that were placed in a perforated tube with several other samples. Tubes were placed in larger thimbles for insertion into the reactor. The samples were loaded and unloaded into the various tubes under water. When they were unloaded, the welded-closed aluminum can was placed directly into a shielded shipping cask. The cask lid was put in place and sealed (or locked down) for transport. As the cask was removed from the basin, it was washed down, placed on a truck, and transported to the Atlanta airport for a return flight to the originator of the sample [Gray 2006]. Because the samples were in welded-closed aluminum cans that were handled under water for loading and unloading, there was little probability for these materials to become airborne and there is no documentation that any loss of material occurred. Therefore, these radionuclides require no discussion of bioassay analysis or internal dosimetry concerns.

Other exotic radionuclides were created in the reactors for various purposes, including the Systems for Nuclear Auxiliary Power – Isotopic Program and the Transplutonium Program. These are discussed in detail in ORAUT-RPRT-0054, *Exotic Radionuclides at Savannah River Site* [ORAUT 2013a]. Research on separation procedures and actual mid-scale or large-scale separations were conducted on some radionuclides in the Savannah River Laboratory (SRL). Most of the exotic radionuclides were monitored by in vivo counting or fission product (FP) urinalysis [ORAUT 2013a, Table 5-1, pp. 21–22]. No bioassay has been found to date for several radionuclides including  $^{99}\text{Tc}$ ,  $^{147}\text{Pm}$ ,  $^{171}\text{Tm}$ ,  $^{249}\text{Bk}$ , and  $^{254}\text{Fm}$ . As discussed in ORAUT-RPRT-0054,  $^{99}\text{Tc}$  was not of dosimetric significance because of the small amount handled combined with its low specific activity. Generally,  $^{147}\text{Pm}$  was contaminated with  $^{144}\text{Ce}$ , which was monitored by WB counting. However, there were two purifications in June 1967 and April 1968 that produced 2.5 kg of essentially pure  $^{147}\text{Pm}$ ; there is no identified evidence that indicated that bioassay was needed [ORAUT 2016]. Chemical activities (purification from irradiated enriched erbium targets) were conducted on large activities of pure  $^{171}\text{Tm}$  at various times from 1966 to 1968. Potential exposure to pure  $^{249}\text{Bk}$  occurred at various times from 1965 through 1972. Cooling of fermium targets before starting of chemical activities precluded exposure to purified  $^{254}\text{Fm}$  (3.2-hr half-life) [ORAUT 2013a].

### 5.3 IN VITRO ANALYSES, MINIMUM DETECTABLE ACTIVITIES, AND COUNTING METHODS

MDAs contained in the site profile are intended as defaults when there is no better information available such as sample-specific MDAs. When the bioassay results in the worker's records include an MDA (or indication that the site considered the value below the MDA, such as "<0.05 dpm/1.5L"), that value takes precedence over the site default value and is to be used in the dose assessment. This applies regardless of whether the sample's less-than value is larger or smaller than the MDA in the site profile.

#### Bioassay Data Results: MDAs and Decision Levels

Since 1990, SRS has used the concepts and formulas from the draft or published radiobioassay standard American National Standards Institute (ANSI)/Health Physics Society (HPS) N13.30, *Performance Criteria for Radiobioassay* [HPS 1996 through HPS 2017<sup>1</sup>], for determination of decision levels and MDAs [WSRC 1990, 1993a, 1994, 1996, 1997, 2001; Savannah River Nuclear Solutions (SRNS) 2008, 2014; Negron 2018; SRNS 2019; Hyman 2019]. At SRS, decision levels were often called "detect levels." The formulas in the 1990 version of the *Internal Dosimetry Technical Basis Manual* [WSRC 1990] varied slightly between some of the radionuclide analyses and from formulas in

<sup>1</sup> The 1989 draft version of ANSI/HPS N13.30 is unavailable. WSRC [1990] indicates SRS had adopted the draft standard before publication.

the 2001 version of the same manual. However, all formulas were based on the concepts of Type I and Type II errors and used 5% for the  $\alpha$  and  $\beta$  probabilities [WSRC 1990, pp. 384–404; 2001, pp. 177–181]. Since 1990, the MDA has been defined as the quantity of material (or activity) that has a 5% chance of not being detected. Before 1990, the term MDA might have had less rigorous or slightly different meanings.

In the early years at SRS (handwritten records), results were reported as an actual value or as a less-than value. Any handwritten results that do not have a “<” sign in front of the reported result should be treated as positive values. Values reported with a “<” sign should be treated as less than the MDA, with the MDA equal to the reported value. For the following example data, many of the uranium results in 1955 and 1956 were reported as actual values rather than less-than (“<”) (censored) values and are shown in Figure 5-1. These results should be treated as positive results.

DATE	ELEMENT	RESULT	LOCATION	
1 8-15-55	Pu	1.05 $\mu$ m/1.5L	200 F	21
2 11-25-55	Pu	.05 $\mu$ m/1.5L	200 F	22
1 7-13-56	U	1.4 $\mu$ g/L	200 F	21
2 7-19-56	U	1.0 $\mu$ g/L	200 F	22
3 7-19-56	F.P	<30 $\mu$ g/250ml	200 F	23
4 8-31-56	Pu	.16 $\mu$ m/1.5L	200 F	24
5 10-15-56	Pu	.52 $\mu$ m/1.5L	200 F	25
6 11-8-56	Pu	.06 $\mu$ m/1.5L	200 F	26
7 12-1-56	Pu	.11 $\mu$ m/1.5L	200 F	27
1 2-19-60	Pu	<1.05 $\mu$ m/1.5L	M	21
2 4-1-60	U	<1 $\mu$ g/L	M	22
3 9-19-60	Pu	5.05 $\mu$ m/1.5L	F	23
4 10-28-60	F.P	130 $\mu$ m/250ml	H	24
1 11-22-60	Pu	1.05 $\mu$ m/1.5L	H	21
2 3-17-61	Pu	5.05 $\mu$ m/1.5L	F	22
3 8-14-61	Pu	5.05 $\mu$ m/1.5L	H	23
4 9-18-61	F.P	460 $\mu$ m/250ml	H	24

Figure 5-1. Examples of positive and reported less-than urinalysis results [ORAUT 2023].

### 5.3.1 Urinalysis

SRS implemented bioassay programs to cover 35 facilities that processed actinides, FPs, and thorium [Thomas et al. 1993].

#### 5.3.1.1 **Routine Sampling Program**

SRS designed the routine sampling program to assess the adequacy of facility controls and personnel protective measures. Bioassay sampling procedures dating back to 1968 detail who would be monitored and for which radionuclides [DuPont 1968]. The routine sampling procedures, such as those in DuPont [1968], emphasized monitoring for those workers with higher exposure potential. In 1965, groups of workers with potential for intakes of tritium and/or FPs were specified. The procedure lists eight work groups by location across several buildings required to leave urine samples for FP analysis [DuPont 1963–1967].

By 1968, SRS designed a category system to identify and track intake potential. Monitoring frequency was based on exposure potential and categories ranged from minimum to maximum potential. Specific nuclides were location dependent, but workers with minimum potential might have been sampled for plutonium, FPs, uranium, or neptunium once every 3 years [DuPont 1968]. Frequency

increased with increasing exposure potential, and the maximum potential groups were sampled as often as four times per year for plutonium and twice per year for FPs. Each category listed specific groups of workers with associated monitoring frequencies. Workers not included in any of the 23 categories were considered not to have a potential for intake and were therefore not sampled. Category definitions and workgroup designations were updated over time in subsequent versions of the bioassay control procedures based on missions and materials, but SRS continued to sample workers by intake potential.

In Revision 3 of the *Bioassay Control* procedure, instructions were provided for collecting samples in the event of suspected inhalations, ingestions, injections, or skin contaminations, or whenever airborne contamination exceeded control guides [DuPont 1970]. In 1971, additional guidance for Construction Division personnel was added but with no specific guidance for trivalent actinides. "Other nuclides," which would have included the trivalent actinides, were monitored as specified by area Health Physics (HP) in the construction Job Plans [DuPont 1971a]. The periodicity of routine urine sampling changed throughout the 1970s for various work locations and as a result of the introduction of in vivo counting [DuPont 1971a,b, 1976–1989].

Logbooks indicate bioassay samples for uranium were analyzed by both fluorophotometric analysis and delayed neutron counting (DNC) in 1982 to 1984. Logbooks also indicate bioassay samples for enriched uranium (EU) were analyzed by both gross alpha analysis and DNC in 1982 to 1985.

The 1990 *Internal Dosimetry Technical Basis Manual* monitoring program for trivalent actinides specified quarterly urine bioassay, an annual chest count, semiannual fecal bioassay, and personal air sampling (PAS) [WSRC 1990]. If monitored by workgroup, SRS decreased the urine bioassay to annual unless a member of the workgroup had a confirmed intake. The routine bioassay program had two parts, prescheduled sampling and job-specific sampling. The program was used in facilities where workers had a reasonable potential for exposure to radioactive materials [WSRC 1990]. A worker's enrollment in the routine bioassay program was based not only on the radioactive hazards associated with the facility where the individual worked, but also on the type of work the worker normally performed. SRS considered a worker who routinely performed tasks requiring respiratory protection to be at greater risk of involvement in an incident resulting in an intake than, for instance, a manager who occasionally performed a cursory walkthrough of a facility.

Therefore, for the purposes of the bioassay program, SRS defined three categories of workers that were in effect through 1993 [WSRC 1993a]:

- Category I. Hands-on workers with the highest intake potential. They routinely work in Contamination or High Contamination Areas and are required to wear respiratory protection. For these employees, the routine program typically requires quarterly urine samples and an annual chest count.
- Category II. Supervisors, engineers, or operators who do not perform hands-on work, but who routinely enter Radiologically Controlled Areas (RCAs) to observe work or record data. They are typically required to submit an annual urine sample but do not receive annual chest counts.
- Category III. Other people working on site who occasionally enter RCAs but only do so to observe work. These workers are at little to no risk of receiving an intake. They are not required to submit routine urine samples, and they do not receive annual chest counts [WSRC 1993b].

With the implementation of 10 C.F.R. Part 835 [10 C.F.R. 835, 1998], SRS defined the three categories of workers for actinide bioassay as:

- Those likely to exceed 100 mrem committed effective dose equivalent (CEDE),<sup>2</sup>
- Those with reasonable potential to exceed 100 mrem CEDE, and
- Those with no potential to exceed 100 mrem CEDE.

From 1953 to 1989 SRS kept bioassay records for each sampled worker on cards. Since 1990, excreta analysis records have been maintained in electronic databases including the Health Protection Radiation Exposure Database (HPRED) and the Health Physics Radiation Exposure Database and Access Control System. The database identifies baseline (new hire), routine, special, and termination measurements.

Because no worker at SRS was considered likely to exceed 100 millirem CEDE [Findley 1998, p. 6], routine sampling for exposure monitoring was not required at SRS during the 1990s either by order or by regulation. Routine sampling was used to confirm the adequacy of workplace monitoring and worker protection programs. For the tritium bioassay program, SRS placed workers on the routine program when they routinely entered RCAs with the potential for tritium contamination or airborne tritium activity [WSRC 1993c, p. 11].

For both actinide and tritium bioassay, prescheduled sampling was used to sample workers who routinely worked in locations with Airborne Radioactivity Area postings. Samples were generally scheduled annually or semiannually based on a worker's date of birth or other recurring event. A large majority of routine bioassay samples were collected with prescheduled sampling. Job-specific sampling was used to sample workers who were not on prescheduled sampling but who needed to enter locations requiring respiratory protection. Like prescheduled sampling, job-specific sampling was used to assess the adequacy of facility controls and personnel protective measures. A definition of the job-specific sampling from September 1997 described it as follows [Findley 1997, p. 8]:

*Job-specific urine sampling is also known as "RWP (Radiation Work Permit) sampling" and is administered exclusively by radiological control personnel. RWPs are written to describe a specific scope of work to be performed. Additional information such as the expected radiological hazards to be encountered, respiratory protection if needed, and dosimetry requirements are listed on the RWP. Based on the degree and nature of the radiological hazards it may be necessary to request each worker performing "hands on" work to leave a urine bioassay sample, hence the term "job-specific sampling." If a worker is wearing respiratory protection and is not on an appropriate routine sampling program then a job-specific sample should be requested.*

Job-specific sampling was not performed to measure a suspected intake. SRS did not assign dose using job-specific bioassay sampling results. However, results of prescheduled and job-specific bioassays could flag a worker for special bioassay sampling. Samples collected from routine sampling for uranium, actinides, and FPs were generally collected over 24 hours, although some were collected as 8-hour samples. Tritium samples were generally single-void samples even when prescheduled.

Starting in 1991, some of the bioassay data reports do not include an MDA or a less-than result but might include information for Result, Activity, and Detect Level as shown in Figure 5-2. The term "detect level" is an SRS term for the decision level. In the early 1990s the reports generally only

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<sup>2</sup> To formalize the definition of who might fit this criterion, SRS performed a statistical study of bioassay data from 1999 and 2000 to test the process of judging a worker's likelihood of exceeding 100 mrem from intakes of radioactive materials [LaBone 2001].



include data in the Result column. If a negative value (i.e., less than zero) is reported in this column, this indicates that SRS considered no activity to be detected and the reported Result is the negative of the Detect Level. When a Result value is reported without the minus sign (as a positive number) and with no associated Detect Level, assume the result is positive (i.e., greater than the decision level). The MDA can be assumed to be twice the decision level [ORAUT 2022]. In the example in Figure 5-2 for  $^{239}\text{Pu}$  on May 7, 1992, the Result is reported as  $-0.020$  dpm/L. With the assumption that the absolute value of this result is the decision level, the MDA is determined to be twice the decision level, in this case  $0.040$  dpm/L, and is normalized to the daily excretion of Reference Man in International Commission on Radiological Protection (ICRP) Publication 23 for the purposes of dose assessment by multiplying by  $1.4$  L/d [ORAUT 2018; ICRP 1975].

Sample ID	Sample Void Date	Sample Receive Date	Facil	Isotope	LLD Ind	Result	Result Units	Re-Run Result	Activity	Detect Level	Error	Sample Type
	12/04/1990	12/18/1990	F	Pu-238		-0.020	dpm/L					ROUTINE
	12/04/1990	12/18/1990	F	Pu-239		-0.010	dpm/L					ROUTINE
	12/01/1991	12/05/1991	F	Pu-238		-0.020	dpm/L					ROUTINE
	12/01/1991	12/05/1991	F	Pu-239		-0.040	dpm/L					ROUTINE
	12/05/1991	12/16/1991	F	Pu-238		-0.030	dpm/L					ROUTINE
	12/05/1991	12/16/1991	F	Pu-239		-0.030	dpm/L					ROUTINE
	05/07/1992	05/13/1992	F	Pu-238		-0.030	dpm/L					ROUTINE
	05/07/1992	05/13/1992	F	Pu-239		-0.020	dpm/L					ROUTINE
	07/21/1992	07/28/1992	F	Pu-238		-0.040	dpm/L					ROUTINE
	07/21/1992	07/28/1992	F	Pu-239		-0.020	dpm/L					ROUTINE
	09/10/1992	09/17/1992	F	Pu-238		-0.030	dpm/L					ROUTINE
	09/10/1992	09/17/1992	F	Pu-239		-0.040	dpm/L					ROUTINE
	12/07/1992	12/10/1992	F	Pu-238		-0.030	dpm/L					ROUTINE
	12/07/1992	12/10/1992	F	Pu-239		-0.030	dpm/L					ROUTINE
	03/04/1993		D	H3	<	0.100		0.000				R
	03/04/1993		D	H3	<	0.100		0.000				R
	03/08/1993	03/09/1993		Pu-238		-0.020	dpm/L					ROUTINE
	03/08/1993	03/09/1993		Pu-239		-0.020	dpm/L					ROUTINE
2111	06/03/1993	06/08/1993		Pu-238		-0.033	dpm/L					ROUTINE
2111	06/03/1993	06/08/1993		Pu-239		-0.033	dpm/L					ROUTINE
23432	05/31/1994	06/07/1994		Pu-238		-0.058	dpm/L		0.033	0.058	0.046	ROUTINE
23432	05/31/1994	06/07/1994		Pu-239		-0.058	dpm/L		-0.006	0.058	0.012	ROUTINE
42161	06/01/1995	06/07/1995		Pu-238		-0.042	dpm/L		-0.011	0.042	0.019	ROUTINE
42161	06/01/1995	06/07/1995		Pu-239		-0.032	dpm/L		-0.006	0.032	0.008	ROUTINE
	06/06/1996	06/10/1996		Pu-238		-0.062	dpm/L		-0.012	0.062	0.033	ROUTINE
	06/06/1996	06/10/1996		Pu-239		-0.039	dpm/L		0.004	0.039	0.019	ROUTINE
74964	06/02/1997	06/09/1997		Pu-238		-0.067	dpm/L		-0.004	0.067	0.040	ROUTINE
74964	06/02/1997	06/09/1997		Pu-239		-0.053	dpm/L		0.005	0.053	0.030	ROUTINE

Figure 5-2. Example 1990s bioassay report from a claims tracking application claim [ORAUT 2023].

Starting in 1994, data for Activity, Detect Level, and Error are usually included along with the Result. In this case, the value reported in the Activity column would be compared to the MDA (twice the Detect Level) to determine if the result would be considered a positive bioassay result. In Figure 5-2, the June 6, 1996,  $^{239}\text{Pu}$  Activity value of  $0.004$  dpm/L would not be considered a positive result because it ( $0.004 \times 1.4 = 0.0056$  dpm/d) is below the MDA, calculated as Detect Level (dpm/L)  $\times 2 \times 1.4$  L/d or  $0.039$  dpm/L  $\times 2 \times 1.4$  L/d =  $0.109$  dpm/d.

Another report format was also used by the site for data beginning in 1991 and is shown with the same bioassay data as discussed with Figure 5-2; an example is given in Figure 5-3. In this example, the Result column shows less-than with a value in the Result column. This value is the decision level.

For example, the reported <sup>239</sup>Pu result for May 7, 1992, is <0.020 dpm/L, as is the case in Figure 5-2. The MDA would be twice this decision level (0.040 dpm/L) and corrected for daily excretion by multiplying by 1.4 L/d (0.056 dpm/d). Note these two reports show the same data in different formats.

Dept	Bottle Date	Rec Date	Type	Batch ID	Isotope	Result	Vol
S12685	12/04/1990	12/18/1990	ROUTINE	01-91-068	Pu-238 <	0.020 dpm/L	839
S12685	12/04/1990	12/18/1990	ROUTINE	01-91-068	Pu-239 <	0.010 dpm/L	839
S12685	12/01/1991	12/05/1991	ROUTINE	13-91-016	Pu-238 <	0.020 dpm/L	889
S12685	12/01/1991	12/05/1991	ROUTINE	13-91-016	Pu-239 <	0.040 dpm/L	889
S1261B	12/05/1991	12/16/1991	ROUTINE	12-91-048	Pu-238 <	0.030 dpm/L	938
S1261B	12/05/1991	12/16/1991	ROUTINE	12-91-048	Pu-239 <	0.030 dpm/L	938
S1261B	05/07/1992	05/13/1992	ROUTINE	05-92-076	Pu-238 <	0.030 dpm/L	1025
S1261B	05/07/1992	05/13/1992	ROUTINE	05-92-076	Pu-239 <	0.020 dpm/L	1025
SF1200	07/21/1992	07/28/1992	ROUTINE	08-92-052	Pu-238 <	0.040 dpm/L	1000
SF1200	07/21/1992	07/28/1992	ROUTINE	08-92-052	Pu-239 <	0.020 dpm/L	1000
SF1200	09/10/1992	09/17/1992	ROUTINE	09-92-124	Pu-238 <	0.030 dpm/L	912
SF1200	09/10/1992	09/17/1992	ROUTINE	09-92-124	Pu-239 <	0.040 dpm/L	912
SF1200	12/07/1992	12/10/1992	ROUTINE	01-93-127	Pu-238 <	0.030 dpm/L	1068
SF1200	12/07/1992	12/10/1992	ROUTINE	01-93-127	Pu-239 <	0.030 dpm/L	1068
SF1200	03/08/1993	03/09/1993	ROUTINE	03-93-089	Pu-238 <	0.020 dpm/L	941
SF1200	03/08/1993	03/09/1993	ROUTINE	03-93-089	Pu-239 <	0.020 dpm/L	941
SF1200	06/03/1993	06/08/1993	ROUTINE	06-93-288	Pu-238 <	0.033 dpm/L	1057
SF1200	06/03/1993	06/08/1993	ROUTINE	06-93-288	Pu-239 <	0.033 dpm/L	1057
SF1224	05/31/1994	06/07/1994	ROUTINE	06-94-112	Pu-238 <	0.058 dpm/L	888
SF1224	05/31/1994	06/07/1994	ROUTINE	06-94-112	Pu-239 <	0.058 dpm/L	888
A3610	06/01/1995	06/07/1995	ROUTINE	06-95-298	Pu-238 <	0.042 dpm/L	1009
A3610	06/01/1995	06/07/1995	ROUTINE	06-95-298	Pu-239 <	0.032 dpm/L	1009
A3630	06/06/1996	06/10/1996	ROUTINE	06-96-278	Pu-238 <	0.062 dpm/L	826
A3630	06/06/1996	06/10/1996	ROUTINE	06-96-278	Pu-239 <	0.039 dpm/L	826
JC3630	06/02/1997	06/09/1997	ROUTINE	06-97-229	Pu-238 <	0.067 dpm/L	974
JC3630	06/02/1997	06/09/1997	ROUTINE	06-97-229	Pu-239 <	0.053 dpm/L	974

Figure 5-3. Example 1990s alternate bioassay report from a claims tracking application claim [ORAUT 2023].

If the code IA (inconclusive analysis) is recorded with a 0.000 in the Result column (Figure 5-2), or 0.000 in the Result column (Figure 5-3) with an IA in the Result Units column, that indicates the analysis did not meet quality control (QC) criteria and no result was reported.

When a report is not in the formats described above and an MDA cannot be determined, MDA tables in this document should be used to determine the MDA.

SRS revised the routine bioassay program in March 1999. SRS provided a Radiological Qualifications Badge to each radiation worker listing each radionuclide of concern other than tritium. HP checked bioassay requirements on badges as a worker signed into an RWP. HP added any radionuclides required by the RWP to the badge if they were not on the badge. Workers delivered their samples to the SRS Whole Body Counting facility during their birth month [Findley 1998].

With implementation of the *U.S. Department of Energy Radiological Control Manual* requirements in 1992 [DOE 1992] and the subsequent regulation 10 CFR Part 835 starting January 1, 1996 [10 C.F.R. 835, DOE 1998], DOE required training before workers could work with radioactive materials:

- Rad Worker I provides appropriate training for individuals who work with radioactive material or radiation producing devices that do not produce high radiation fields.
- Rad Worker II includes all of the training for Rad Worker I plus additional training appropriate for individuals who might enter high radiation or contaminated areas or who might work with unsealed quantities of radioactive materials.

All workers with potential exposure to unsealed sources of radioactive materials must have completed Rad Worker II training. Rad Worker II personnel were also analyzed by WB counting and when required by chest counting during a worker's birth month [Findley 1998].

The MDAs are shown for each radionuclide in tables in Sections 5.3 and 5.4. These values are applicable over the specified periods. The lack of an MDA in a given period reflects the lack of documented information about urinalysis for that radionuclide. Activity fractions for depleted uranium (DU), EU, and natural uranium (NU) are listed in the tables in Section 5.3.2.4.

The following codes are found in the excreta records:

- C Routine urine sample including baselines and termination samples;
- A-A 24-hour special urine sample;
- A-O Two 24-hour urine samples, one analyzed, one held;
- A-W Follow-up to positive routine sample;
- IA Inconclusive Analysis, analysis did not meet QC criteria and no result reported;
- B Follow-up to an IA sample;
- A-F Single void fecal samples;
- T Tritium sample, only 50 mL collected; and
- P-10 An old code for tritium.

MDA values since about 1988 are twice the average decision level for the analysis. These MDAs apply to both routine and special urinalyses. Since about 1990, routine analysis results (except tritium) that exceed the decision level are recounted for twice the normal count time. The recounted result is reported instead of the original count. Some samples are "rerun," meaning a second aliquot is

processed. In these cases, the bioassay results show two results with different sample numbers for the same collection date and same sample volume.

### 5.3.1.2 Special Bioassay Program

SRS designed the Special Bioassay program to assess “inadvertent intakes” of radioactive material that could exceed the 100-mrem threshold [LaBone 2001] to comply with 10 C.F.R. § 835.432(c) starting on January 1, 1996 [10 C.F.R. 835, 1998]. Under this program, samples were required in response to unusual or unanticipated circumstances. For example, a sample was required whenever a worker was suspected of receiving an intake that would result in a CEDE of 100 mrem or greater [LaBone 2001, p. 2]. Intakes of radionuclides were assigned when positive results of special bioassay sampling were confirmed [SRS no date].

### 5.3.1.3 Double-Blind Quality Assurance Program

SRS implemented a program to submit “double-blind” spiked QC samples into the routine in vitro sample stream in 1999 [WSRC 1999]. The program, which is still in use, is designed to allow the internal dosimetry group to independently assess the timeliness and accuracy of the analytical laboratory [WSRC 2006a]. The analytical laboratory knows neither that such samples were spiked nor the activity level (thus making a double-blind sample). To ensure that the samples appear to be valid, the QCs are submitted with an employee identifier.

Results from analysis of double-blind samples might or might not be reported in the records. From November 2003 to September 2012, all double-blind samples were assigned to the same staff member of HP. Details on that staff member are in Priester [2019]. For the periods before and after November 2003 to September 2012, the samples were assigned to SRS workers not on the routine bioassay program [SRNS 2013a]. Starting in 1999 when evaluating urinalysis results for  $^{238/239}\text{Pu}$ ,  $^{237}\text{Np}$ , or  $^{241}\text{Am}$ , dose reconstructors should check Sample Type. The result is from a double-blind QC sample if the Sample Type is QC. As needed, contact the author of this TBD section for further investigation if both results of both radionuclides are positive.

Results from double-blind sampling should not be included in dose assessment.

## 5.3.2 Analytical Methods for Individual Radionuclides

The following section discusses the in vitro bioassay methods for specific radionuclides. General bioassay requirements by area and building are given in ORAUT-TKBS-0003-7, *Savannah River Site – Internal Dosimetry Co-Exposure Data* [ORAUT 2024b, Table 7-18]. MDAs are provided for most radioanalyses; consult Section 5.7.2 when the MDA is questionable.

### 5.3.2.1 Trivalent Actinides (Americium, Curium, and Californium)

Records showing urinalysis for trivalent actinides date back at least to the mid-1960s using liquid ion exchange: tri-isooctylamine (TIOA) followed by di-2-ethylhexyl phosphoric acid [Butler 1964, p. 4]. The MDA for this method was about 0.3 dpm  $\pm$  0.3 dpm (90% confidence limit) of either  $^{235}\text{U}$  or Am-Cm, which can be detected after a 2-hour count [Butler 1964, p. 10]. On or about 1970, an extraction method using the bidentate dibutyl N,N-diethylcarbonylphosphonate was developed that allowed sequential separation of plutonium, neptunium, and uranium with TIOA, followed by extraction of thorium, americium, curium, berkelium, californium, and einsteinium with bidentate. The sensitivity of that method was reported to be 0.02  $\pm$  0.01 dpm/250 mL or 0.12 dpm/1.5 L for a 24-hr count [Butler and Hall 1970, pp. 2, 4]. In Revision 3 of the *Bioassay Control* procedure [DuPont 1970], the positive level for total activity from trivalent actinides (americium, curium, and californium) was noted as 0.3 dpm/1.5 L and the sample positive level was used for the resample level. The sample size was

reduced to 250 mL. An intake was considered confirmed if the initial bioassay result was >1 dpm/1.5 L and a resample result was 0.3 dpm/1.5 L [DuPont 1970, p. 2]. Alpha spectrometry was used to identify individual radionuclides, but the sensitivity appeared to be based on a gross alpha count [Butler and Hall 1970, pp. 3–4]. In 1971, the MDA using gross alpha counting on a solid-state detector was listed as 0.3 dpm/1.5 L [Taylor 2000, p. 4]. In 1990, SRS began using radiochemical processing (ion exchange resin) with an MDA of 0.15 dpm/L [WSRC 2001, p. 182; Taylor et al. 1995, p. 79]. Alpha spectrometry has been used since 1992 for special samples and since 1995 is the only method used for routine samples with MDAs of 0.064 dpm/L for <sup>241</sup>Am and 0.047 dpm/L for <sup>244</sup>Cm and <sup>252</sup>Cf [WSRC 2001, p. 58]. A review of the recorded data shows that the transition from gross alpha to alpha spectrometry was not clean; a few routine samples had alpha spectrometry in 1993 and 1994. Before implementation of alpha spectrometry, SRS reported gross alpha results as AmCmCf. MDA values reported by SRS from 1964 forward are summarized in Table 5-1.

Table 5-1. AmCmCf in vitro urinalysis MDAs.

Nuclide	Period	MDA	Method	Sources
Am-241	1964–1967	0.3 dpm/500 mL	Liquid ion exchange	Butler 1964, pp. 9–10
Am-241	1968–1969	1 dpm/250 mL	Liquid ion exchange	DuPont 1968, p. 2
Am-241	1970–1989	0.3 dpm/1.5 L	TIOA extraction	DuPont 1970, p. 2; 1971a, p. 2; 1971b, p. 2; 1976–1989, pp. 2–3; DuPont/WSRC 1967–1993, pp. 118, 274
Am-241	1990–1994	0.2 pCi/L	Ion exchange resin	WSRC 1990, pp. 199, 396; 1992, p. 33; 1993a, p. 48; 1994, p. 55
Am-241	1995–2007	0.029 pCi/L	Extraction chromatography resin	WSRC 1996, p. 62; 1997, p. 59; 2001, p. 58
Am-241	2008–2011	0.025 pCi/L	Extraction chromatography resin	SRNS 2008, p. 66
Am-241	2012–present <sup>a</sup>	0.019 pCi/L	Extraction chromatography resin	SRNS 2012, p. 75; 2014, p. 74; 2019, p. 46; 2020, p. 44
Am-243	1992–1995	0.2 pCi/L	Extraction chromatography resin	WSRC 1990, pp. 199, 396; 1992, p. 33; 1993a, p. 48; 1994, p. 55
Am-243	1996–2011	Not applicable	Extraction chromatography resin	WSRC 1996, p. 62; 1997, p. 59; 2001, p. 58; SRNS 2008, p. 66
Am-243	2012–present <sup>a</sup>	0.032 pCi/L	Extraction chromatography resin	WSRC 1996, p. 62; 1997, p. 59; 2001, p. 58; SRNS 2008, p. 66
Cm-242	1990–1995	0.2 pCi/L	Extraction chromatography resin	WSRC 1990, pp. 199, 396; 1992, p. 33; 1993a, p. 48; 1994, p. 55
Cm-242	1996–present <sup>a</sup>	Not applicable	Extraction chromatography resin	WSRC 1996, p. 62; 1997, p. 59; 2001, p. 58; SRNS 2008, p. 66; 2012, p. 75; 2014, p. 74; 2019, p. 46; 2020, p. 44
Cm-244	1964–1967	0.3 dpm/500 mL	Liquid ion exchange	Butler 1964, p. 10
Cm-244	1968–1969	1 dpm/250 mL	Liquid ion exchange	DuPont 1968, p. 2
Cm-244 <sup>a</sup>	1970–1989 <sup>a</sup>	0.3 dpm/1.5 L	TIOA extraction	DuPont 1970, p. 2; 1971a, p. 2; 1971b, p. 2; 1976–1989, pp. 2–3; DuPont/WSRC 1967–1993, pp. 127, 274
Cm-244	1990–1994	0.2 pCi/L	Extraction chromatography resin	WSRC 1990, pp. 199, 396; 1992, p. 33; 1993a, p. 48; 1994, p. 55
Cm-244	1995–2007	0.021 pCi/L	Extraction chromatography resin	WSRC 1996, p. 62; 1997, p. 59; 2001, p. 58
Cm-244	2008–2011	0.02 pCi/L	Extraction chromatography resin	WSRC 1996, p. 62; 1997, p. 59; 2001, p. 58; SRNS 2008, p. 66
Cm-244	2012–2018	0.013 pCi/L	Extraction chromatography resin	SRNS 2008, p. 66
Cm-244	2019–present <sup>a</sup>	0.023 pCi/L	Extraction chromatography resin	SRNS 2019, p. 46; 2020, p. 44
Cf-252	1970–1989	0.3 dpm/1.5 L	TIOA extraction	DuPont 1970, p. 2; 1971a, p. 2; 1971b, p. 2; 1976–1989, pp. 2–3; DuPont/WSRC 1967–1993, pp. 127–274
Cf-252	1990–1994	0.2 pCi/L	Ion exchange resin	WSRC 1990, pp. 199, 396; 1992, p. 33; 1993a, p. 48; 1994, p. 55
Cf-252	1995–2007	0.021 pCi/L	Extraction chromatography resin	WSRC 1996, p. 62; 1997, p. 59; 2001, p. 58
Cf-252	2008–2011	0.02 pCi/L	Extraction chromatography resin	WSRC 1996, p. 62; 1997, p. 59; 2001, p. 58; SRNS 2008, p. 66

Nuclide	Period	MDA	Method	Sources
Cf-252	2012–2018	0.014 pCi/L	Extraction chromatography resin	WSRC 1996, p. 62; 1997, p. 59; 2001, p. 58; SRNS 2008, p. 66
Cf-252	2019–present <sup>a</sup>	0.023 pCi/L	Extraction chromatography resin	SRNS 2019, p. 46; 2020, p. 44

a. The 2020 SRS *Internal Dosimetry Technical Basis Manual* is the latest source. Apply MDA listed in SRNS [2020].

Use of a gross alpha count before 1994 means that the measurement itself does not indicate which radionuclide or combination of radionuclides was measured. Location or other information about an intake might allow determination of the principal radionuclide. If not, an assumption that the material is <sup>241</sup>Am is favorable to the claimant. Exposure to einsteinium, berkelium, or fermium was possible in the SRL from 1964 through the 1970s [ORAUT 2013a, pp. 38–42].

### 5.3.2.2 Plutonium

From the beginning of the plutonium urinalysis program in 1954 to approximately 1959, urine samples were radiochemically processed using bismuth phosphate and lanthanum fluoride coprecipitation and electroplated. Activities were determined by gross alpha track analysis of exposed nuclear track emulsion, Type A (NTA). In 1959, SRS replaced the bismuth phosphate method with nitric acid/hydrogen peroxide dissolution through ion exchange, which was faster and used less volume of urine but had essentially the same MDA [Sanders and Leidt 1961]. From about 1965 to 1988, counting for gross alpha activity was performed using a solid-state surface barrier alpha detector. TIOA liquid extraction replaced the ion exchange chemistry in 1966. This method also used direct evaporation on planchets instead of electrodeposition, and it allowed separation of neptunium and uranium from the same sample. Sensitivity was stated to be 0.1 dpm/1.5 L. In or about 1981, SRS introduced a new coprecipitation technique for routine samples along with alpha spectrometry. SRS also introduced a sample-specific determination of plutonium recovery by use of a <sup>242</sup>Pu tracer at that time. Results for <sup>238</sup>Pu and <sup>239</sup>Pu were reported separately. The TIOA method with gross alpha counting continued to be used on special samples until 1988 [Taylor 2000, p. 3]. SRS introduced an electronic database in 1990; results were thereafter reported per liter. Electrodeposition was reinstated in 1994. Separation of plutonium with neptunium, actinides, uranium, and strontium from a single sample using a trioctyl and tridecyl methyl ammonium-based resin (TEVA) and transuranic (TRU) resins began in 2001 [WSRC 2001, p. 169]. Alpha-emitting plutonium and neptunium isotopes are electrodeposited and counted by alpha spectrometry on a single planchet. Tables 5-2 and 5-3 provide a history of plutonium in vitro bioassay MDAs.

SRS used the following equation to derive MDA in WSRC [1990, p. 389]:

$$MDA = \frac{[4.65(S_{blk}) + 3](Unit\ Correction\ Factor)}{Time \times Counter\ Efficiency \times Recovery \times Volume} \quad (5-1)$$

where

<i>MDA</i>	=	minimum detectable activity (pCi/L)
<i>S<sub>blk</sub></i>	=	standard deviation (SD) of the count from a set of acceptable blanks (counts)
<i>Unit Correction Factor</i>	=	converts dpm to desired activity units (e.g., conversion to pCi would be 0.45 pCi/dpm)
<i>Time</i>	=	sample count time (min)
<i>Counter Efficiency</i>	=	efficiency for radionuclide on specific counter (cpd)
<i>Recovery</i>	=	recovery (unitless)
<i>Volume</i>	=	amount of sample analyzed (L)

Table 5-2. <sup>239/240</sup>Pu in vitro urinalysis MDAs.

Period	MDA	Method	Sources
1954–1958	0.08 dpm/1.5 L	Bismuth phosphate, lanthanum fluoride coprecipitation, gross alpha NTA film track counting	Sanders 1956, p. 13
1959–1965	0.084 dpm/1.5 L	Ion exchange (start 1959), gross alpha NTA film track counting to 1964 with solid-state alpha spectroscopy counting after 1964	Sanders and Leidt 1961, pp. 6–7
1966–1988	0.084 dpm/1.5 L	Coprecipitation, electrodeposition, solid-state alpha spectroscopy (routine samples); liquid TIOA extraction, electrodeposition gross alpha (special samples)	Butler 1968, p. 5
1989–1991	0.02 pCi/L	Ion exchange followed by electrodeposition and solid-state alpha spectroscopy for Pu-239/240	MDA: WSRC 1990, pp. 164–166 (0.02 pCi/L); Method: WSRC 1990, pp. 365–366
1992–1993	0.04 pCi/L	Ion exchange followed by electrodeposition and solid-state alpha spectroscopy for Pu-239/240	MDA: WSRC 1992, pp. 18–20; 1993a, pp. 35–36; Method: WSRC 1992, p. 15; 1993a, p. 31
1994–1995	0.015 pCi/L	Ion exchange followed by electrodeposition and alpha spectroscopy for Pu-239/240	MDA: WSRC 1994, p. 42; Method: WSRC 1994, p. 38
1996–2000	0.022 pCi/L	Coprecipitation, wet ash/cleanup, ion exchange column, electrodeposition, and solid-state alpha spectroscopy for Pu-239/240	MDA: WSRC 1996, p. 50; 1997, pp. 49–50; Method: WSRC 1996, p. 198; 1997, p. 184
2001–2007	0.022 pCi/L	Coprecipitation, TEVA/TRU stacked columns, wet ash/cleanup, TEVA column, electrodeposition followed by solid-state alpha spectroscopy for Pu-239/240	MDA: WSRC 2001, p. 58; Method: WSRC 2001, p. 182
2008–2011	0.015 pCi/L	Coprecipitation, TEVA/TRU stacked columns, wet ash/cleanup, TEVA column, electrodeposition followed by solid-state alpha spectroscopy for Pu-239/240	MDA: SRNS 2008, pp. 55–57, 66; Method: SRNS 2008, pp. 75, 143
2012–2018	0.015 pCi/L	Coprecipitation, TEVA/TRU stacked columns, wet ash/cleanup, TEVA column, electrodeposition followed by solid-state alpha spectroscopy for Pu-239/240	MDA: SRNS 2012, pp. 66, 75; 2014, pp. 66, 74; Method: SRNS 2012, pp. 62, 154; 2014, pp. 61, 153
2019–present <sup>a</sup>	0.022 pCi/L	Coprecipitation, TEVA/TRU stacked columns, wet ash/cleanup, TEVA column, electrodeposition followed by solid-state alpha spectroscopy for Pu-239/240	MDA: SRNS 2019, pp. 39, 46; 2020, p. 44; Method: SRNS 2019, p. 73; 2020, p. 72

a. The 2020 SRS *Internal Dosimetry Technical Basis Manual* is the latest source. Apply MDA listed in SRNS [2020].

Table 5-3. <sup>238</sup>Pu in vitro urinalysis MDAs.

Period	MDA	Method	Sources
1954–1958	0.08 dpm/1.5 L	Bismuth phosphate, lanthanum fluoride coprecipitation, gross alpha NTA film track counting	Sanders 1956, p. 13
1959–1965	0.084 dpm/1.5 L	Ion exchange (start 1959), gross alpha NTA film track counting to 1964 with solid-state alpha spectroscopy counting after 1964	Sanders and Leidt 1961, pp. 6–7
1966–1988	0.084 dpm/1.5 L	Coprecipitation, electrodeposition, solid-state alpha spectroscopy (routine samples); liquid TIOA extraction, electrodeposition gross alpha (special samples)	Butler 1968, p. 5
1989–1991	0.02 pCi/L	Ion exchange followed by electrodeposition and solid-state alpha spectroscopy for Pu-238	MDA: WSRC 1990, pp. 164–166 (0.02 pCi/L), p. 396 (0.015 pCi/L); Method: WSRC 1990, pp. 365–366
1992–1993	0.03 pCi/L	Ion exchange followed by electrodeposition and solid-state alpha spectroscopy for Pu-238	MDA: WSRC 1992, pp. 18–20; 1993a, pp. 35–36; Method: WSRC 1992, p. 15; 1993a, p. 31
1994–1995	0.018 pCi/L	Ion exchange followed by electrodeposition and solid-state alpha spectroscopy for Pu-238	MDA: WSRC 1994, p. 42; Method: WSRC 1994, p. 38
1996–2000	0.032 pCi/L	Coprecipitation, wet ash/cleanup, ion exchange column, and electrodeposition solid-state alpha spectroscopy for Pu-238	MDA: WSRC 1996, p. 50; 1997, pp. 49–50; Method: WSRC 1996, p. 198; 1997, p. 184
2001–2007	0.032 pCi/L	Coprecipitation, TEVA/TRU stacked columns, wet ash/cleanup, TEVA column, electrodeposition followed by solid-state alpha spectroscopy for Pu-238	MDA: WSRC 2001, p. 58; Method: WSRC 2001, p. 182
2008–2011	0.016 pCi/L	Coprecipitation, TEVA/TRU stacked columns, wet ash/cleanup, TEVA column, electrodeposition followed by solid-state alpha spectroscopy for Pu-238	MDA: SRNS 2008, pp. 55–57, 66 (note that the MDA is listed as 0.016 on p. 57 and 0.030 on p. 66); Method: SRNS 2008, pp. 75, 143
2012–2018	0.016 pCi/L	Coprecipitation, TEVA/TRU stacked columns, wet ash/cleanup, TEVA column, electrodeposition followed by solid-state alpha spectroscopy for Pu-238	MDA: SRNS 2012, pp. 66, 75; 2014, pp. 66, 74; Method: SRNS 2012, pp. 62, 154; 2014, pp. 61, 153
2019–present <sup>a</sup>	0.023 pCi/L	Coprecipitation, TEVA/TRU stacked columns, wet ash/cleanup, TEVA column, electrodeposition followed by solid-state alpha spectroscopy for Pu-238	MDA: SRNS 2019, pp. 39, 46; SRNS 2020, p. 44; Method: SRNS 2019, p. 73; SRNS 2020, p. 72

a. The 2020 SRS *Internal Dosimetry Technical Basis Manual* is the latest source. Apply MDA listed in SRNS [2020].



SRS reported a revision to the MDA equation in 2001 [WSRC 2001, p. 177].

$$MDA = \frac{[4.65 \times \sqrt{C_{bkg}} + 3]}{TREVA} \quad (5-2)$$

where

- MDA** = minimum detectable activity (pCi/L)  
 **$C_{bkg}$**  = SD in the count position background count; given as the square root of the background counts in the region of interest  
 **$T$**  = sample count time (min)  
 **$R$**  = percent recovery  
 **$E$**  = average counter efficiency (cpm/dpm)  
 **$V$**  = amount of sample analyzed (L)  
 **$A$**  = abundance, radioisotope alpha

Plutonium existed in mixtures of  $^{238}\text{Pu}$ ,  $^{239}\text{Pu}$ ,  $^{240}\text{Pu}$ , and  $^{241}\text{Pu}$ . The relative activities of these isotopes depended on the nature of the irradiation of the uranium fuel (referred to as burnup) and the time between the end of irradiation and the intake. Processes at SRS did not disrupt the relative activities of the isotopes. In general, SRS created plutonium mixtures ranging from about 3 wt%  $^{240}\text{Pu}$  to about 12 wt%  $^{240}\text{Pu}$ . SRS usually blended the plutonium mixtures to produce a final product with about 6 wt%  $^{240}\text{Pu}$ , referred to as a weapons-grade mixture. The  $^{241}\text{Pu}$ , a beta emitter with a 14-yr half-life, undergoes significant decay to  $^{241}\text{Am}$  during the lifetime of the facilities, so the isotopic ratios of contamination will be different from the ratios in newly irradiated fuel or feed to the Head End Stream. Americium-241 builds up from near zero at end of irradiation; however, it is removed during production of the plutonium product, then begins to build up again as the  $^{241}\text{Pu}$  remaining in the product decays. Therefore, the ratio of  $^{241}\text{Pu}$  to, for example,  $^{239+240}\text{Pu}$  decreases from the time of end of irradiation, whereas the ratio of  $^{241}\text{Am}$  to  $^{239+240}\text{Pu}$  increases from the time of last separation of the  $^{241}\text{Am}$  from the plutonium.

Many different mixtures of plutonium isotopes were possible in the FB-Line, but it is assumed that the mixture was either 6 wt% or 12 wt% plutonium. The percent refers to the mass percent of  $^{240}\text{Pu}$ . The radionuclides of concern in Table 5-4 are the same for both internal and external exposure. The table provides activity compositions for reference 6 wt% and 12 wt%  $^{240}\text{Pu}$  mixtures for specific activities and activity ratios. Fresh plutonium applies to the period from 1954 to 1958, 5-year-aged plutonium applies to the period from 1959 to 1963, and 10-year-aged plutonium applies to 1964 and later.

Table 5-4. Activity composition for reference 6 wt% and 12 wt%  $^{240}\text{Pu}$  mixtures. <sup>a,b</sup>

Mixture designation, years of aging (right) and isotope (below)	Specific activities (Ci/g)					
	6 wt%, fresh <sup>c</sup>	6 wt%, 5-yr <sup>d</sup>	6 wt%, 10-yr <sup>d</sup>	12 wt%, fresh <sup>c</sup>	12 wt%, 5-yr <sup>d</sup>	12 wt%, 10-yr <sup>d</sup>
Pu-238	8.56E-03	8.23E-03	7.91E-03	1.71E-02	1.64E-02	1.58E-02
Pu-239+240	7.13E-02	7.13E-02	7.13E-02	7.98E-02	7.98E-02	7.98E-02
Pu-241	8.24E-01	6.48E-01	5.09E-01	3.09E+00	2.43E+00	1.91E+00
Am-241	0.00E+00	5.83E-03	1.04E-02	0.00E+00	2.19E-02	3.89E-02
Pu alpha	7.99E-02	7.95E-02	7.92E-02	9.69E-02	9.62E-02	9.56E-02
Total alpha	7.99E-02	8.53E-02	8.96E-02	9.69E-02	1.18E-01	1.35E-01

## Activity ratios

Mixture designation, years of aging (right) and isotope (below)	6 wt%, fresh <sup>c</sup>	6 wt%, 5-yr <sup>d</sup>	6 wt%, 10-yr <sup>d</sup>	12 wt%, fresh <sup>c</sup>	12 wt%, 5-yr <sup>d</sup>	12 wt%, 10-yr <sup>d</sup>
Pu239+240:Am-241	N/A	1.22E+01	6.87E+00	N/A	3.64E+00	2.05E+00
Pu-alpha:Am-241	N/A	1.36E+01	7.61E+00	N/A	4.39E+00	2.46E+00
Pu-241:Pu-239+240	1.16E+01	9.09E+00	7.14E+00	3.87E+01	3.05E+01	2.39E+01
Pu-241:Pu-alpha	1.03E+01	8.15E+00	6.43E+00	3.18E+01	2.53E+01	2.00E+01
Pu-238:Pu 239+240	1.20E-01	1.15E-01	1.11E-01	2.14E-01	2.06E-01	1.98E-01
Pu-238:Pu-alpha	1.07E-01	1.04E-01	9.99E-01	1.76E-01	1.70E-01	1.65E-01

- a. Source: Carbaugh [2003, pp. 183–184].  
b. N/A = not applicable.  
c. Two weeks after end of irradiation.  
d. Time since separation of the Am-241.

**Plutonium Mix – Fresh (6 wt% and 12 wt%)**

A ratio of <sup>241</sup>Am ingrowth for 2 weeks after separation is used based on Tables 8-3 and 8-4 of *Methods and Models of the Hanford Internal Dosimetry Program* [Carbaugh 2003, pp. 183–184]. This ratio should also be used for americium in the fresh (2-week-aged) material.

The ratio (6 wt% and 12 wt%) for fresh (2-week aged) material is included in the SRS Pu-Mix Calculator and the SRS Pu-Mix Intake Rate Calculator tools, which are available in the SRS dose reconstruction tools folder. Plutonium bioassay results before 1981 are not the same quantity as results in 1981 and after because of the counting techniques. Before 1981 the result was for total alpha from the plutonium isotopes (i.e., both <sup>238</sup>Pu and <sup>239</sup>Pu). Starting in 1981, the <sup>238</sup>Pu results were counted and reported separately from the <sup>239</sup>Pu results. The text provides two ways to normalize the results before curve fitting or intake calculations. However, higher burnup material was made as part of the curium campaigns (Table 5-5). The latter material had a higher than normal weight percentage of <sup>242</sup>Pu, which was used to produce curium, which in turn was irradiated to produce <sup>252</sup>Cf. This program ran from 1963 to 1967 [DuPont 1967; Bebbington 1990] and would have been an exposure potential for workers in the canyon facilities and 235-F [DuPont 1965a].

Table 5-5. Specific activity (Ci/g) composition and ratios for high <sup>242</sup>Pu mixture.<sup>a</sup>

Mixture designation/years of aging (right) <sup>a</sup> and isotope (below)	Specific activity		
	6 wt% Pu, fresh <sup>b</sup>	6 wt% Pu, 5 yr <sup>c</sup>	6 wt% Pu, 10 yr <sup>c</sup>
Pu-238	0.00E+00	0.00E+00	0.00E+00
Pu-239+240	1.87E-02	1.87E-02	1.87E-02
Pu-241	3.87E-03	3.04E-03	2.39E-03
Pu-242	3.34E-03	3.34E-03	3.34E-03
Am-241	0.00E+00	2.74E-05	4.88E-05
Pu alpha	2.20E-02	2.20E-02	2.20E-02
Total alpha	2.20E-02	2.20E-02	2.21E-02

## Activity ratios

Mixture designation/years of aging (right) <sup>a</sup> and isotope (below)	6 wt% Pu, fresh <sup>b</sup>	6 wt% Pu, 5 yr <sup>c</sup>	6 wt% Pu, 10 yr <sup>c</sup>
Pu-239+240:Am-241	Not applicable	6.80E+02	3.83E+02
Pu-alpha:Am-241	Not applicable	8.03E+02	4.52E+02
Pu-241:Pu-239+240	2.07E-01	1.63E-01	1.28E-01
Pu-241:Pu-alpha	1.76E-01	1.38E-01	1.08E-01
Pu-242:Pu-239+240	1.79E-01	1.79E-01	1.79E-01
Pu-242:Pu-alpha	1.52E-01	1.52E-01	1.52E-01

a. Source: DuPont [1965a, p. 52] except as noted.

b. Two weeks after end of irradiation. Calculated using specific activities and half-lives from ICRP Report 107 [ICRP 2008].

c. Time since separation of the <sup>241</sup>Am.

### 5.3.2.3 Tritium

From 1954 until 1958, SRS analyzed tritiated water (HTO) vapor in urine by passing hydrogen evolved from the urine sample through an ionization chamber; the reported MDA for this method was 1 µCi/L. In 1958, SRS initiated liquid scintillation counting, which is still in use. The sample size was listed as one voiding in 1968 with a positive level of 1 µCi/L and a resample level of 5 µCi/L [DuPont 1968, p. 2]. Taylor et al. [1995, p. 7] reports that the MDA consistently improved to the current level of 20,000 pCi/L (or 0.02 µCi/L). The reporting level remained at the value of 1 µCi/L, was later reduced to 0.5 µCi/L on or about October 1976 [DuPont 1976–1989, p. 3], and then reduced further to the current value of 0.1 µCi/L (an exact date was not determined, but at least by 1989). During the 1980s, although the reporting level of 0.5 µCi/L was generally used, some results below 0.5 are listed (e.g., 0.4, 0.3). Considering that the true MDA was probably well below the reporting level, these results should be considered positive.

Note that within the HPRED (1990 to 2003), tritium results of 0.05 µCi/L or less are reported as <0.1µCi/L and results between 0.05 µCi/L and 0.1 µCi/L are reported as 0.1 µCi/L.

Tritium analyses are listed as T on the employee bioassay cards. Tritium might also be listed as P-10, especially in the 1950s. Tritium results in the 1990s were listed on the same summary form as external dose monitoring results. They are referred to as sample results with dates and analysis results, but the word tritium or any other radionuclide identifier is not mentioned per se. For tritium results, the denominator used for reporting purposes has always been per liter of urine. (The denominator of 1.5 L was never used for tritium as it was for other radionuclides.) Only tritium bioassay results greater than 5 µCi/L were manually evaluated before computer evaluation of data. SRS identified reporting levels and MDAs for tritium through 2019 (Table 5-6).

Table 5-6. Tritium in vitro urinalysis MDAs ( $\mu\text{C/L}$ ).

Period	MDA	Reporting level	Sources
1953–1975	1.0	1 $\mu\text{C/L}$	MDA and reporting level [Taylor et al. 1995, p. 78]; MDA [DuPont 1968, p. 2; 1970, p. 2]; [DuPont 1971a, p. 2; 1971b, p. 2]
1976–1986	1.0	0.5–1.0 $\mu\text{C/L}$	MDA and reporting level [Taylor et al. 1995, p. 78]; MDA [DuPont 1976–1989, p. 3]; [DuPont/WSRC 1967–1993, p. 274]
1987–1989	0.15	0.5 $\mu\text{C/L}$	MDA and reporting level [Taylor et al. 1995, p. 78]; [DuPont/WSRC 1967–1993, pp. 127, p. 118]
1990–2007	0.02	0.1 $\mu\text{C/L}$	MDA and reporting level [Taylor et al. 1995, p. 78]; MDA [WSRC 1990, p. 198]; [WSRC 2001, p. 57]; reporting level [WSRC 1990, p. 368; 2001, p. 185]
2008–2018	0.017	0.1 $\mu\text{C/L}$	MDA [SRNS 2008, p. 65; 2014, p. 73]; reporting level [SRNS 2008, p. 160; 2014, p. 170]
2019–present	0.007	Not applicable	MDA [SRNS 2019, p. 46; 2020, p. 44]

### 5.3.2.4 Uranium

SRS has used a variety of methods historically to analyze uranium in vitro at SRS. These methods and the associated detection capabilities are summarized in Table 5-7. Use values as shown in the individual's records if available. Isotopic parameters and ratios for EU, DU, NU, and target residual material (TRM) are listed in Tables 5-8 through 5-11.

#### Occupational Uranium Intakes

Based on information in ORAUT-OTIB-0060, *Internal Dose Reconstruction*, any result other than a baseline sample that is positive (as defined in ORAUT-OTIB-0060 Section 2.4.2 and further detailed in Section 2.4.3 for occupational intakes of uranium) is assumed to be due to an occupational exposure for the purposes of dose reconstruction [ORAUT 2018].

#### Uranium Enrichment Assumptions

When the work area is not known and no uranium source term is available, the following assumptions should be made for uranium exposure when only mass-based urinalysis results are available to convert units of mass to units of activity [McCarty 2000]. For 1953 to 1967, assume NU at 0.6928 pCi/ $\mu\text{g}$ ). For 1968 and later, assume DU at 0.372 pCi/ $\mu\text{g}$ .

#### Uranium-233

Uranium-233 was created by irradiation of thorium slugs and extracted in the H Canyon buildings in several campaigns from 1964 to 1969. The bioassay would have been the EU method with alpha counting because of the high specific activity of  $^{233}\text{U}$ . Uranium-232 was an impurity in  $^{233}\text{U}$ , but its alpha activity would have been included in the bioassay results. The  $^{228}\text{Th}$  progeny of  $^{232}\text{U}$  (half-life 1.9 yr) would have built up slowly in the final product, but this is accounted for by assuming intakes of  $^{232}\text{Th}$  and progeny for the canyon building. Potential thorium intakes should not be considered through September 1972 [Sebelius 2012, p. 3]. There were FP impurities in the  $^{233}\text{U}$  product as well, but they were listed only as gamma counts per minute per mL of solution in the monthly technical reports [DuPont 1965a,b, 1984a,b]. A similar program was conducted at Hanford [ORAUT 2020a, pp. 42–43], so the nonuranium impurities were assumed to be similar. Uranium-233 was primarily a radionuclide of interest in the locations listed by period in Table 5-25 later in this document.

Table 5-7. Uranium urinalysis MDAs.<sup>a</sup>

Period	Uranium mixture	MDA	Method	Sources
Startup–mid-1960s	EU <sup>a</sup>	0.15 dpm/1.5 L	Gross alpha for uranium, alpha track counting	WSRC 1995, pp. 78–79
Mid-1960s–1982	EU	1 dpm/1.5 L	Gross alpha for uranium on solid-state detector	WSRC 1995, pp. 78–79
1954–1982	DU	5 µg/1.5L	Fluorophotometric analysis	WSRC 2001, pp. 184–185; Mann 1962, p. 2
1983–1986	U-235	0.14 ng	DNC for U-235	WSRC 1995, pp. 78–79
1983–1986	NU	1 µg/L	DNC for U-235	WSRC 1995, pp. 78–79
1983–1986	EU	1 dpm/L	DNC for U-235	WSRC 1995, pp. 78–79
1987–1989	EU	1 dpm/1.5 L	Gross alpha for uranium on solid-state detector	WSRC 1995, pp. 78–79
1986 <sup>a</sup> –1989	DU	1 µg/L	Kinetic phosphorimetry analysis	WSRC 2001, pp. 184–185; DuPont/WSRC 1967–1993, p. 147
1990–1994 <sup>b</sup>	Total U	0.4 pCi/L	Gross alpha for uranium on solid-state detector	WSRC 1990, pp. 169–172
1990–1991	U-234	0.2 pCi/L	Alpha spectroscopy for specific uranium isotopes	WSRC 1990, pp. 368, 396
1990–1991	U-235	0.2 pCi/L	Alpha spectroscopy for specific uranium isotopes	WSRC 1990 pp. 368, 396
1990–1991	U-236	0.2 pCi/L	Alpha spectroscopy for specific uranium isotopes	WSRC 1990, pp. 368, 396
1990–1991	U-238	0.2 pCi/L	Alpha spectroscopy for specific uranium isotopes	WSRC 1990, pp. 368, 396
1992–1995	U-234	0.2 pCi/L	Alpha spectroscopy for specific uranium isotopes	WSRC 1992, pp. 33, 44; 1994, pp. 55, 191
1992–1995	U-235	0.2 pCi/L	Alpha spectroscopy for specific uranium isotopes	WSRC 1992, pp. 33, 44; 1994, pp. 55, 191
1992–1995	U-236	0.2 pCi/L	Alpha spectroscopy for specific uranium isotopes	WSRC 1992, pp. 33, 44; 1994, pp. 55, 191
1992–1995	U-238	0.2 pCi/L	Alpha spectroscopy for specific uranium isotopes	WSRC 1992, pp. 33, 44; 1994, pp. 55, 191
1996–2000	U-234	0.18 pCi/L	Alpha spectroscopy for specific uranium isotopes	WSRC 1996, pp. 62, 211
1996–2000	U-235	0.22 pCi/L	Alpha spectroscopy for specific uranium isotopes	WSRC 1996, pp. 62, 211
1996–2000	U-238	0.18 pCi/L	Alpha spectroscopy for specific uranium isotopes	WSRC 1996, pp. 62, 211
2001–2007	U-234	0.032 pCi/L	Alpha spectroscopy for specific uranium isotopes	WSRC 2001, pp. 58, 173
2001–2007	U-235	0.036 pCi/L	Alpha spectroscopy for specific uranium isotopes	WSRC 2001, pp. 58, 173
2001–2007	U-238	0.032 pCi/L	Alpha spectroscopy for specific uranium isotopes	WSRC 2001, pp. 58, 173
2008–2011	U-234	0.024 pCi/L	Alpha spectroscopy for specific uranium isotopes	SRNS 2008, pp. 66, 149
2008–2011	U-235	0.026 pCi/L	Alpha spectroscopy for specific uranium isotopes	SRNS 2008, pp. 66, 149
2008–2011	U-238	0.024 pCi/L	Alpha spectroscopy for specific uranium isotopes	SRNS 2008, pp. 66, 149
2012–2018	U-234	0.017 pCi/L	Alpha spectroscopy for specific uranium isotopes	SRNS 2012, pp. 75, 160; 2014 pp. 74, 159
2012–2018	U-235	0.018 pCi/L	Alpha spectroscopy for specific uranium isotopes	SRNS 2012, pp. 75, 160; 2014 pp. 74, 159
2012–2018	U-238	0.016 pCi/L	Alpha spectroscopy for specific uranium isotopes	SRNS 2012, pp. 75, 160; 2014 pp. 74, 159
2019–present	U-234	0.091 pCi/L	Alpha spectroscopy for specific uranium isotopes	SRNS 2019, pp. 46, 79; 2020 pp. 44, 76
2019–present	U-235	0.095 pCi/L	Alpha spectroscopy for specific uranium isotopes	SRNS 2019, pp. 46, 79; 2020 pp. 44, 76
2019–present	U-238	0.095 pCi/L	Alpha spectroscopy for specific uranium isotopes	SRNS 2019, pp. 46, 79; 2020 pp. 44, 76

a. In early bioassay records sometimes labeled En. Uranium or LMF.

b. Claimant records indicate that both gross alpha and isotopic analysis occurred from 1990 to 1994.

Table 5-8. Radiological parameters of DU.<sup>a</sup>

Nuclide	wt%	Specific constituent activity in mixture (μCi/g, nCi/mg, or pCi/μg)	Specific constituent activity in mixture (dpm/μg)	Constituent fraction of total uranium activity in mixture
U-234	0.0005	0.0312	0.0694	0.084
U-235	0.2496	0.0054	0.0120	0.0145
U-236	0.00	0.0000	0.0000	0
U-238	99.7499	0.3354	0.7445	0.9015
<b>Totals</b>	<b>100</b>	<b>0.3720</b>	<b>0.8259</b>	<b>1</b>

a. Sources: SRNS [2020, p. 40].

Table 5-9. Radiological parameters of NU.<sup>a</sup>

Nuclide	wt%	Specific constituent activity in mixture (μCi/g, nCi/mg, or pCi/μg)	Specific constituent activity in mixture (dpm/μg)	Constituent fraction of total uranium activity in mixture
U-234	0.0055	0.3434	0.7624	0.4957
U-235	0.7214	0.0156	0.0346	0.0225
U-236	0.00	0.0000	0.0000	0
U-238	99.2731	0.3338	0.7410	0.4818
<b>Totals</b>	<b>100</b>	<b>0.6928</b>	<b>1.5379</b>	<b>1</b>

a. Sources: SRNS [2020, p. 40].

Table 5-10. Radiological parameters of EU.<sup>a</sup>

Nuclide	wt%	Specific constituent activity in mixture (μCi/g, nCi/mg, or pCi/μg)	Specific constituent activity in mixture (dpm/μg)	Constituent fraction of total uranium activity in mixture
U-234	1.5626	97.2591	215.9152	0.8489
U-235	63.6265	1.3748	3.0522	0.012
U-236	24.59	15.9024	35.3034	0.1388
U-238	10.2230	0.0344	0.0763	0.0003
<b>Totals</b>	<b>100</b>	<b>114.5708</b>	<b>254.3471</b>	<b>1</b>

a. Sources: SRNS [2020, p. 40].

Table 5-11. Radiological parameters of TRM.<sup>a,b</sup>

Nuclide	wt%	Specific constituent activity in mixture (μCi/g, nCi/mg, or pCi/μg)	Specific constituent activity in mixture (dpm/μg)	Constituent fraction of total uranium activity in mixture
U-234	0.0000017	0.0001	0.0002	0.0003
U-235	0.2712	0.0059	0.0130	0.0167
U-236	0.02	0.0100	0.0221	0.0284
U-238	99.7134	0.3353	0.7443	0.95547
<b>Totals</b>	<b>100</b>	<b>0.3512</b>	<b>0.7796</b>	<b>1.0009</b>

a. Sources: SRNS [2020, p. 40].

b. TRM was a term for uranium remaining from extractions of <sup>99</sup>Mo irradiated targets from a non-SRS facility.

## Uranium-236

Uranium-236 is an artificial isotope that can be found in spent nuclear fuel or in reprocessed uranium. The presence of this isotope in a sample of uranium is evidence that the sample has been in a nuclear reactor. Uranium-236 is formed in a nuclear reactor from the fissile isotope <sup>235</sup>U. Most absorption reactions result in fission reaction, but some reactions result in radiative capture of neutrons forming <sup>236</sup>U. About 85% of all absorption reactions result in fission as shown in Figure 5-4. MDA values reported for <sup>236</sup>U by SRS are in Table 5-7.

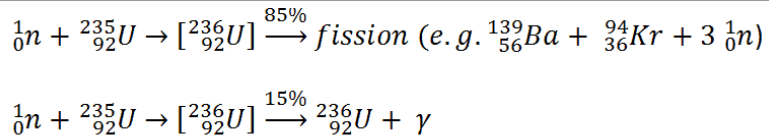


Figure 5-4.  ${}^{235}\text{U}$  to  ${}^{236}\text{U}$  reaction [Uranium 235 consumption, no date].

Uranium-236 has a half-life of  $2.34 \times 10^7$  years, longer than any other artificial actinide or FP produced in nuclear reactors. Uranium-236 has about 190 times higher specific activity than the isotope  ${}^{238}\text{U}$ . This results in high contribution to radioactivity of reprocessed uranium. The amount of  ${}^{236}\text{U}$  present in spent nuclear fuel is directly proportional to the burnup levels and initial  ${}^{235}\text{U}$  enrichment. For discharged H-modified fuel with about 50 MWd/tU enriched to 4%, the  ${}^{236}\text{U}$  typically represents about 0.5 wt%. Uranium-236 decays via alpha decay to  ${}^{232}\text{Th}$ .

Uranium-236 is fissionable only by fast neutrons. Radiative capture of a neutron leads to the formation of the isotope  ${}^{237}\text{U}$ , which quickly beta decays to the isotope  ${}^{237}\text{Np}$ . As such, SRS irradiated  ${}^{236}\text{U}$  to form  ${}^{237}\text{Np}$ .

Isotopic distributions of  ${}^{236}\text{U}$  in uranium were reported at 6.45 wt% in Reactors and H area and 6.4 wt% in A Area [Centers for Disease Control and Prevention 2006, p. 116]. Isotopic distribution of 21 wt% was reported for  ${}^{236}\text{U}$  during the Building 321-M deinventory casting campaign [DOE 1996]. The reported activity fraction of  ${}^{236}\text{U}$  in EU is in Table 5-10 in Section 5.3.2.4.

### 5.3.2.5 Polonium-210

SRS first discussed potential separation of  ${}^{210}\text{Po}$  with the U.S. Atomic Energy (AEC) Commission Savannah River Operations Office in the early 1960s. Details on the separation and use of  ${}^{210}\text{Po}$  at SRS are provided in ORAUT-RPRT-0048, *Feasibility of Reconstructing Doses Received from Potential Exposures to Polonium at Savannah River Site* [ORAUT 2011]. Rather than following the development of  ${}^{210}\text{Po}$  heat sources, the development of  ${}^{238}\text{Pu}$  heat sources were pursued.

Separation of polonium from irradiated bismuth metal was performed in 1967 in a laboratory-scale effort with up to four trials and one documented trial in 1968. Personnel who worked with separated  ${}^{210}\text{Po}$  in Building 773-A were monitored by bioassay in February 1967, which coincides with the first bismuth irradiation event of 1967 performed a few days before. Personnel were also monitored in September and November 1967. Results of all polonium bioassay results were reported as less than <20 dpm/1.5 L on February 7, 1967, and <1 dpm/1.5 L on August 21, 1967. SRS workers with the potential for internal intake of polonium were monitored by bioassay. SRS HP monitored locations where polonium was used with no events of contamination highlighted in monthly reports [ORAUT 2011].

### 5.3.2.6 Strontium-90 and Other Fission Products

Hanford developed a procedure for FP urinalysis in May 1950 [Thorburn 1950]. It is likely that the SRS procedure was similar, at least through 1960. The procedure separated and counted radionuclides of strontium, yttrium, cerium, and promethium. FP analyses were coded FP, FPIA, or IA under ELEMENT on the worker's bioassay cards. The IA code can have two meanings depending on the context. An IA code with a result before 1960 most likely refers to the FP analysis; an IA code with no result means inconclusive analysis. That is, the laboratory was unable to produce a valid result. This analysis was a gross beta analysis and did not account for radionuclides of ruthenium or cesium. DuPont specified bioassay operating guides, sampling frequencies, instructions for requesting and collecting urine samples, and related administrative controls in the *Bioassay Control* procedures. The

earliest available version of the procedure is Revision 2 dated January 2, 1968 [DuPont 1968]. It indicates an FP sample size of 500 mL was used with a positive result level of 100 dpm/1.5 L and a resample level of 200 dpm/1.5 L. The sample request process for 24-hour composite samples required approval by an HP Senior Supervisor or above, indicating that routine samples were probably not 24-hour samples. The bioassay cards show less-than values that varied over the years and probably indicate a switch from beta counting to photon counting; for example, the less-than value was <30 dpm/750 mL from 1955 to 1960, <100 dpm/1.5 L from 1960 to 1965, and <1 nCi/1.5 L from 1966 to 1969. Some records in the 1960 to 1965 period show two results for the same sample: one for beta (<100 dpm/1.5 L) and one for gamma (<500 dpm/1.5 L). MDA values reported by SRS from 1953 forward for FP are in Table 5-12.

Table 5-12. FP urinalysis MDAs.

Period	MDA	Method	Sources
1953–06/1962	30 dpm/750 mL	Coprecipitation followed by beta Geiger-Müller counting	MDA: Boni 1959, p. 10; Method: Taylor 2000, p. 5
07/1962–1967	100 dpm/1.5 L beta, 500 dpm/1.5 L gamma	Coprecipitation followed by beta Geiger-Müller counting	MDA: Mann 1962, p. 2; Method: Taylor 2000, p. 5
1968–1989	0.1 nCi/1.5 L beta, 1 nCi/1.5 L gamma	Liquid ion exchange followed by beta proportional counting	MDA: DuPont 1968, p. 2; Method: Taylor 2000, p. 5
1990–1995	6 pCi/L	Liquid ion exchange followed by beta proportional counting	MDA: WSRC 1990, pp. 192, 199; 1992, pp. 31, 33; 1993a, pp. 45, 48; 1994, pp. 52, 55; Method: Taylor 2000, p. 5
1996–2000	2.9 pCi/L	Ion exchange method modified to extract Sr-90 followed by beta proportional counting	MDA: WSRC 1996, pp 60, 62; Method: Taylor 2000, p. 5
2001–2018	3.9 pCi/L	Ion exchange method modified to extract Sr-90 followed by beta proportional counting	MDA: WSRC 2001, pp. 56, 58, 180; SRNS 2008, pp. 66, 156; 2012, pp. 75, 74; Method: WSRC 2001, pp. 173, 179; SRNS 2014, p. 74; 2008, pp. 146, 155; 2012, pp. 157, 166; 2014, pp. 156, 165
2019–present	3.46 pCi/L	Ion exchange method modified to extract Sr-90 followed by beta proportional counting	MDA: SRNS 2019, p. 45; Method: SRNS 2019, pp. 76, 84; 2020, pp. 43, 73, 82

From the startup of the reactors in December 1953 until 1969, strontium was separated by alkaline earth phosphate coprecipitation followed by beta counting on a Geiger-Müller or proportional counter [WSRC 1990, p. 376]. Urine samples were acidified with nitric and orthophosphoric acid and a cobalt carrier solution. Ammonium hydroxide was added, and the FPs precipitated. The precipitate was fired dry, dissolved with nitric acid, transferred to a planchet, dried again, and counted. This analysis was also called FP analysis. Recovery was greater than 90% for strontium, yttrium, and cerium/promethium radioisotopes. The lower limit of sensitivity was 29 dpm/750 mL for  $^{90}\text{Sr}/\text{Y}$  [Boni 1959, p. 10; Mann 1962, p. 3]. Both  $^{89}\text{Sr}$  and  $^{90}\text{Sr}$  would have been counted (and potentially radioisotopes of cerium and promethium).

Taylor [2000] states that from 1969 to 1995 strontium was analyzed by liquid ion exchange that separates the yttrium progeny followed by beta proportional counting. Yttrium-91 would be included as a possible interference but  $^{89}\text{Sr}$  would not. Yttrium-91, grown in from  $^{91}\text{Sr}$ , might also be present at intake if the source of intake was recent FPs (up to about a year postirradiation).

Since 2001,  $^{90}\text{Sr}$  has been separated as part of the use of TRU and TEVA resins for plutonium, uranium, actinides, and strontium. MDA values reported by SRS from 1953 forward are in Table 5-12.



### 5.3.2.7 Neptunium-237

Revision 8 of the *Savannah River Site Internal Dosimetry Technical Basis Manual* states that neptunium analyses started in the 1950s, perhaps 1959, using coprecipitation with calcium-magnesium ammonium phosphate and ion exchange followed by autoradiography on NTA for 10,000 minutes [WSRC 2001]. The MDA was not given but was mostly likely similar to the MDA for plutonium alpha autoradiography method used at the same time, i.e., about 0.035 dpm/L. The results were recorded as “S.P.” (special product). Analysis by TIOA extraction and electrodeposition occurred at the same time as for plutonium analyses (about 1964), which was a faster method. DuPont specified bioassay operating guides, sampling frequencies, instructions for requesting and collecting urine samples, and related administrative controls in the *Bioassay Control* procedures. The earliest available version of the procedure is Revision 2 dated January 2, 1968 [DuPont 1968]. It indicates a neptunium sample size of 250 mL was used with a positive result level of 0.1 dpm/1.5 L and a resample level of 0.5 dpm/1.5 L (the same as for plutonium). The sample request process indicates that 24-hour composite samples required approval by an HP Senior Supervisor or above, which suggests that routine samples were probably not 24-hour samples.

In Revision 3 of the *Bioassay Control* procedure in 1970, the positive level for neptunium was noted as 0.1 dpm/1.5 L and the positive level was used for the resample level (also the same as for plutonium) [DuPont 1970]. An intake was considered confirmed if the initial bioassay results were >0.5 dpm/1.5 L and a resample was >0.1 dpm/1.5 L. Neptunium was no longer part of the routine sampling program but was sampled when requested by area HP, and it was stated that “area Health Physics will provide Personnel Monitoring with a list of employees requiring neptunium analysis [NP] if plutonium urinalysis is positive” [DuPont 1970].

Throughout the 1970s, the sample collection guidance for neptunium remained the same in that samples were only collected from personnel designated by area HP when plutonium urinalysis samples were positive [DuPont 1971a,b, 1976–1989]. The 1990 *Internal Dosimetry Technical Basis Manual* monitoring program for TRU elements other than plutonium, which includes neptunium, specified a worker monitoring program of quarterly urine bioassay, an annual chest count, semiannual fecal bioassay, and PAS. If monitored by workgroup, annual urine bioassay and an annual chest count were specified [WSRC 1990]. From 1993 to 1996, samples were analyzed by an offsite commercial laboratory using extraction chromatography resin. In 1996, SRS introduced on site alpha spectrometry. Since 2001, TEVA and TRU resins are used to separate neptunium from actinides and uranium, and alpha spectrometry has continued. Determine the MDA based on the information on the report (i.e., twice reported decision level) or use the MDAs listed in the table. MDA values reported by SRS from 1959 forward are in Table 5-13.

Table 5-13. <sup>237</sup>Np in vitro urinalysis MDAs.

Period	MDA	Sources
~1959–1961	0.05 dpm/1.5 L	MDA: for Pu, Np-237 values are likely similar Mann 1962, p. 3; Taylor 2000, p. 3
1962–1965	0.1 dpm/1.5 L	MDA: for Pu, Np-237 values are likely similar Mann 1962, p. 2; Taylor 2000, p. 3
1966–1989	0.1 dpm/1.5 L	MDA: for Pu, Np-237 values are likely similar DuPont 1968, p. 2; 1970, p. 2; 1971a, p. 2; 1971b, p. 2; 1976–1989, pp. 2–3
1990–1995	0.2 pCi/1 L	MDA: WSRC 1990, p. 396; 1992, pp. 21, 33; 1993a, pp. 37, 48; 1994, pp. 44, 55; Method: WSRC 1990, pp. 371–372; 1992, pp. 44–45; 1993a, pp. 166–167; 1994, pp. 182–184
1996–2000	0.029 pCi/L	MDA: WSRC 1996, pp. 52, 62; 1997, pp. 52, 59; Method: WSRC 1996, pp. 200–201; 1997, pp. 186–188
2001–2011	0.035 pCi/L	MDA: WSRC 2001, pp. 50, 58; SRNS 2008, pp. 55–56, 66; Method: WSRC 2001, p. 169; SRNS 2008, p. 143
2012–2018	0.017 pCi/L	MDA: SRNS 2012, pp. 67, 75; 2014, pp. 66, 74; Method: SRNS 2012, p. 154; 2014, p. 153
2019–present	0.028 pCi/L	MDA: SRNS 2019, pp. 40, 46; 2020, pp. 38, 44; Method: SRNS 2019, p. 73; 2020, pp. 69–70

### 5.3.2.8 Thorium

Due to the infeasibility of reconstructing doses from thorium for the first SEC period, bioassay methods to reconstruct doses from thorium are not included. NIOSH has not identified bioassay techniques directly used to detect intakes of thorium at SRS from October 1, 1972, through about 1989. However, radiochemical procedures used to extract actinides in biological samples also extracted any thorium activity starting in 1970 [Butler and Hall 1970]. The authors state that all alpha-emitting actinides from thorium through einsteinium were extracted giving an excellent gross alpha analytical method. Data from the research showed that, in the analysis of americium, curium, and californium, any contaminating plutonium, neptunium, or uranium had to be removed. However, SRS found that thorium, berkelium, and einsteinium were not present in sufficient quantities to require separation. This procedure is discussed in ORAUT-TKBS-0003-7 [ORAUT 2024b]. SRS used this separation procedure through 1989.

MDA values reported by SRS from 1990 forward are in Table 5-14.

Table 5-14. Thorium in vitro urinalysis MDAs.

Nuclide	Period	MDA	Sources
Th-228	1990–1995	0.1 pCi/L	WSRC 1990, p. 199; 1992, p. 33; 1993a, p. 48; 1994, p. 55
Th-228	1996–2011	Not applicable	WSRC 1996, p. 62; 1997, p. 59; 2001, p. 58; 2008, p. 66
Th-228	2012–present	0.02 pCi/L	SRNS 2012, p. 75; 2014, p. 74; 2019, p. 46; 2020, p. 44
Th-232	1990–1995	0.1 pCi/L	WSRC 1990, p. 199; 1992, p. 33; 1993a, p. 48; 1994, p. 55
Th-232	1996–2011	Not applicable	WSRC 1996, p. 62; 1997, p. 59; 2001, p. 58; 2008, p. 66
Th-232	2012–present	0.013 pCi/L	SRNS 2012, p. 75; 2014, p. 74; 2019, p. 46; 2020, p. 44

### 5.3.3 Fecal Sample Analysis

SRS technical documentation indicates that fecal sample analysis has been performed periodically since the 1950s. These samples were always for special, rather than routine, purposes and were always performed in conjunction with urine bioassay sampling. Fecal samplings were single void samples and reported as dpm/sample.

The history of the use of fecal sampling at SRS is not contiguous. The TIOA procedure for urinalysis was modified to separate and analyze fecal samples in the 1970s, but it is uncertain if the fecal procedure was used and for how long. In the 1980s, fecal samples were analyzed by gamma spectroscopy using a phoswich detector [WSRC 1997, p 191]. MDAs are not available from before those reported in WSRC [1990]. Fecal samples were processed by an offsite vendor from the late 1980s to 1993. The vendor provided MDAs for alpha-emitting isotopes as 0.1 pCi per sample and for  $^{89/90}\text{Sr}$  as 20 pCi/sample [WSRC 1990, p 159, WSRC 1997, p 191]. Starting in 1994, fecal samples are screened by photon counting with the following typical MDAs: 7 pCi ( $^{241}\text{Am}$ ), 300 pCi ( $^{238}\text{Pu}$ ), and 600 pCi (weapons-grade plutonium) [WSRC 1997, pp. 206–207]. However, if results could not be quantified by photon counting, that is, if nothing was detected, the samples might have been analyzed by normal separation chemistry and alpha spectrometry to improve the detection capability. This protocol applies mostly to samples from 1994 to the present and probably was not applied to samples before 1994. Fecal results that are associated with normal separation chemistry likely did not capture the plutonium activity associated with Type SS material. Fecal results that are associated with dissolution and gross counting of ash, with no normal separation chemistry, should capture the plutonium activity associated with Type SS material. By 2000, SRS collected fecal samples for some special bioassay investigations. Those samples are processed and analyzed on site using the same procedures used for liquid samples, and fecal samples have not been collected without a corresponding urine sample. Fecal samples are reported as activity per day [WSRC 2001, p. 68]. The fecal analysis MDA is assumed to be the same as the corresponding urine sample [WSRC 2001, p 140].

## 5.4 IN VIVO BIOASSAY MINIMUM DETECTABLE ACTIVITIES, COUNTING METHODS, AND REPORTING PRACTICES

SRS has used both WB counting and chest counting to detect internally deposited intakes of radionuclides.

### 5.4.1 Whole Body Counting

WB counting began in approximately 1960, using a 4-in.-high by 8-in.-thick diameter NaI(Tl) detector. The detection energy range was 100 keV to 2,000 keV. The counting room was built of 12-in.-thick pre-World War II steel measuring 8 ft wide by 11 ft long by 8 ft high. The monitored individual sat in a reclining chair positioned in an arc around the detector; this was referred to as the “40-cm arc geometry” in bioassay monitoring reports. SRS used a Remcal phantom to calibrate the counter. Bioassay via this method was not used for plutonium and americium due to their low-energy emissions. MDA values for 1960 to October 1974 are in Table 5-15.

Table 5-15. WB counting MDAs, 1960 to October 1974 (nCi).<sup>a,b</sup>

Nuclide	MDA <sup>c</sup>
Ce-144	29
U <sup>d</sup>	62
I-131	1.4
Ru-106	6.1
Cs-137	1.0
Zr/Nb-95	2.2
Zn-65	5.1
Ba/La-140	9.3
Co-60	0.3 <sup>e</sup>

- Source: Taylor et al. [1995, Table 11].
- Counting times are not documented in source.
- MDA method:  $2.899 \times \text{SD}$  from average clean person.
- Listed as uranium but is NU.
- Watts and Wright [1965–1972, p. 224], based on a calculation made in November 1971.

In the early 1970s, WB counting using an array of NaI(Tl) detectors was implemented. The MDA values for November 1974 to December 1979 are in Table 5-16.

Table 5-16. WB counting MDAs, November 1974 to December 1979 (nCi).<sup>a,b,c</sup>

Nuclide	MDA
Ce-144	13
I-131	5 <sup>d</sup>
Ru-106	12
Cs-137	10
Zr/Nb-95	3
Zn-65	9
Co-60	3

- Source: Hall [1967–1979, p. 162].
- These values may be used for 1980 to 1987 if no count-specific MDA values are provided.
- Counting times are not documented in source.

- d. I-131 MDA obtained with counts in approximate position of thyroid [Hall 1967–1979, p. 111].

Many forms for reporting the results of WB counts were used throughout the years. The reporting format used from the onset of WB counting until about 1974 is shown in Figure 5-5; that reporting format did not list MDAs. That form has column headings “nuclide,” “energy,” “B.G. c/m,” “40 cm Arc net c/m,” and “results.” The first column and the results column are the columns of concern; the other columns involve preliminary steps. The  $^{137}\text{Cs}$  results are reported as “ncCs,” meaning nCi of  $^{137}\text{Cs}$ , and “pcCs/gmK,” apparently meaning pCi of  $^{137}\text{Cs}/\text{g}$  of potassium calculated from the  $^{40}\text{K}$  result. The dose reconstructor only needs the ncCs value. The letters A, B, C, and D under the nuclide column appear to be placeholders for that energy region of the spectrum; there was not a single radionuclide of interest that had a peak in that region.

If MDAs are not shown in the worker’s records, use Table 5-15 to 5-18. A 10-nCi followup level for  $^{137}\text{Cs}$  was implemented sometime before 1975, which might show up as an MDA. However, this was based on counts on nonoccupationally exposed persons and was not reflective of the minimum sensitivity of the detector system [Hall 1967–1979, p. 96].

An example of the earliest WB count reporting format used from the onset of counting through the 1970s is shown in Figure 5-5.

The earliest forms used to report WB count results from the onset of chest counting to late 1970s are shown in Figures 5-6 and 5-7.

The reporting format used from about 1979 to 1986 (an example is shown in Figure 5-8), which also reported chest count results, listed the MDA for each radionuclide for each count (i.e., count-specific MDAs). This form has two main blocks of data, one under “WBC (NAI)” referring to the WB count results, the other under “Chest Count (NAI)” referring to the results from the chest counter. The columns list the nuclide and other intermediate steps in the calculation, and the last column gives the MDA for the nuclide, meaning the count did not detect the nuclide above the MDA. If there is no value in the last column, then there should be a special listing just below the columns. This is where the result in nCi is shown for a nuclide that is detected. Dose reconstructors should expect  $^{40}\text{K}$  to be listed in this location as a minimum.

In the late 1980s, SRS bought several commercial standup WB counters (Canberra FASTSCAN) with large NaI(Tl) detectors in a shadow shield arrangement for counting high-energy photons. SRS mounted those counters in trucks and used them as mobile counters, primarily at F Area and Central Shops. In 1986, a new form to report WB count results was used with the new Canberra counters. An example of this form, which was used until 1994, is shown in Figure 5-9.

SRS opened the new In-Vivo Counting Facility in September 1995 [SRNS 2008]. The building, 735-4B, has 3,400 ft<sup>2</sup> and is divided into three steel-shielded counting rooms, a control room and corridor, a stand-up FASTSCAN WB counter in an unshielded room, three change rooms, and four offices. SRS implemented a new combination scanning bed WB and chest counter at this time. The scanning bed counter is configured with four low- or broad-energy germanium detectors, one NaI(Tl) detector, a motor-controlled scanning (MCS) unit, and a horizontal subject counting bed. The low- or broad-energy detectors are used for chest counting, and the NaI(Tl) detector and MCS unit are used for WB counting. The chest counting detectors are mounted on an adjustable support arm, which allows for proper placement of the detector faces above the counting subject. The WB detector is a single 3- by 5- by 16-in. NaI(Tl) detector mounted on a movable belt beneath the centerline of the horizontal counting bed. The movement of the NaI(Tl) detector is controlled by the MCS unit. The MCS unit is useful in determining the approximate location in the body of fission and activation

product activity detected by the NaI(Tl) detector. The format used to report WB counts is shown in Figure 5-10.

MDAs for these counting systems were generated individually for counts by processing software.

In 1990 SRS listed new MDA values using the following equation [WSRC 1990, p. 431]:

$$MDA = \frac{2.71 + 4.66 \times \left[ \sum_{c-w}^{c+w} Y_i \right]^{1/2}}{t_i \times E \times @} \quad (5-3)$$

where

<i>MDA</i>	=	minimum detectable activity (pCi/L)
<i>c</i>	=	centroid energy
<i>w</i>	=	user variable window function, set at 0.64*full-width half maximum (FWHM)
<i>Y<sub>i</sub></i>	=	number of counts in gross spectrum
<i>t<sub>i</sub></i>	=	count time
<i>E</i>	=	detection efficiency at centroid energy
@	=	photon yield

**WHOLE BODY COUNTER DATA**

Date: 2/13/62 Time: 1245

Name: [Redacted] P.R.No. [Redacted] Dept.: Reactor Bldg.: 452

Height: 66 1/2 Weight: 135 1/2 Sex: [Redacted] Age: [Redacted]

SRP Radiation Worker  SRP Non-Radiation Worker  New Employee  Visitor

Counting Time: \_\_\_\_\_ Energy Calibration: \_\_\_\_\_  
 If not otherwise indicated = 20 minutes If not otherwise indicated = 10 Kev/Ch.

Background Tape attached to Data Sheet of: [Redacted]

Nuclide	Energy Mev.	B.G. c/m	40 cm Arc Net c/m	Results	
Nd <sup>147</sup>	.06-.10	81.9	37.6		
Ce <sup>144</sup>	.07-.14	119.8	63.3		
U	.15-.22	127.5	51.4		
Np <sup>237</sup>	.23-.29	102.6	37.6		
I <sup>131</sup>	.30-.40	116.1	39.2		
Ru <sup>106</sup>	.45-.55	89.2	23.9		
Cs <sup>137</sup>	.60-.70	56.2	45.0	47.8 $\frac{\mu\text{Ci}}{\text{gm}}$	5.77 $\frac{\text{mCi}}{\text{kg}}$
Zr <sup>95</sup>	.70-.80	52.1	13.8		
"A"	.81-.91	41.8	10.8		
"B"	.92-1.02	35.4	6.3		
Zn <sup>65</sup>	1.03-1.17	36.8	11.7		
"C"	1.18-1.29	26.5	10.7		
K <sup>40</sup>	1.30-1.54	55.2	53.6	1.96 $\frac{\text{gmK}}{\text{Kg b.w.}}$	121 gmK
Ba <sup>140</sup>	1.55-1.75	23.7	5.5		
"D"	1.76-1.99	18.3	0.4		

MTH-202

Figure 5-5. Example of early whole body count form, February 13, 1962 [ORAUT 2023].

DATE 11/7/78  
 TIME 1735  
 NAME  
 PR#  
 LOCN 232H  
 DEPT SEP  
 AGE  
 SEX  
 REQU ROUTINE

HT 71.25  
 WT 184  
 C THKS 2.0

**Whole body**

CH#	6-10	7-14	15-22	23-29	30-40	45-55	60-70
GROS	+ 2885	+14111	+18536	+15217	+12458	+14748	+ 8881
BKGD	+ 2885	+12548	+16353	+14237	+11403	+13021	+ 8492
NET	+ 2885	+ 1561	+ 2183	+ 978	+ 1054	+ 1726	+ 389
CALC		+ 1539	+ 2442	+ 1846	+ 1087	+ 1024	+ 669
DIFF		+ 22	- 259	- 867	- 32	+ 701	- 280

CH#	71-80	81-91	92-102	103-117	118-129	130-154	155-175	176-199
GROS	+ 4506	+ 4422	+ 3780	+ 4151	+ 2454	+ 6191	+ 2431	+ 1767
BKGD	+ 4116	+ 3990	+ 3174	+ 3690	+ 2272	+ 4314	+ 2172	+ 1436
NET	+ 389	+ 432	+ 607	+ 461	+ 182	+ 1877	+ 259	+ 330
CALC	+ 373	+ 340	+ 341	+ 463	+ 229			
DIFF	+ 16	+ 91	+ 265	- 2	- 47			

CH#	29-45	46-96	97-133	134-180	181-256
GROS	+ 356	+ 1259	+ 1322	+ 1547	+ 1676
BKGD	+ 180	+ 462	+ 449	+ 574	+ 572
NET	+ 177	+ 797	+ 873	+ 972	+ 1105
CALC	+ 137	+ 829	+ 823	+ 1007	
DIFF	+ 39	- 32	+ 49	- 34	

**Chest**

GM K + 187  
 NCI CS-137 + 3.70

**Whole body results**

MDA  
 PU-239 + 9 NCI  
 PU-238 + 4 NCI  
 CM-244 + 5 NCI  
 AM-241 + .12 NCI  
 EU + 6 NCI

**Chest count results**

Figure 5-6. Example of early in vivo bioassay form, with both WB and chest count results, used through late 1970s, November 7, 1978 [ORAUT 2023].

DATE 3-29-77  
TIME 01730

LOC. 221-H  
DEPT. SEP.  
AGE  
SEX  
ROUTINE  
HT. 71.25  
WT. 182  
CT. 2.5

**Chest count spectrum**

GROSS C/2000S	BKGD	NET C/30M
START=0029 STOP=0045 INT=0000000373	-160 = 213 x 9 = 192	-74
START=0046 STOP=0096 INT=0000001320	-314 = 1006 x 9 = 905	-81
START=0097 STOP=0133 INT=0000001226	-297 = 929 x 9 = 836	-85
START=0134 STOP=0188 INT=0000001609	-375 = 1234 x 9 = 1111	-106
START=0181 STOP=0256 INT=0000001762	-427 = 1335 x 9 = 1202	

*5 KEV CH.*

START=2056 STOP=2063 INT=0000018538	-16079 = 2459 x 9 = 2213	-18
START=2064 STOP=2071 INT=0000015380	-13535 = 1845 x 9 = 1661	-18
START=2072 STOP=2078 INT=0000012707	-11635 = 1072 x 9 = 965	-7
START=2079 STOP=2089 INT=0000015402	-14548 = 854 x 9 = 769	-6
START=2094 STOP=2104 INT=0000009444	-8876 = 568 x 9 = 511	-4
START=2109 STOP=2119 INT=0000007368	-6956 = 412 x 9 = 371	
START=2120 STOP=2129 INT=0000004690	-4437 = 253 x 9 = 228	-2
START=2130 STOP=2140 INT=0000004769	-4333 = 436 x 9 = 392	-3
START=2141 STOP=2151 INT=0000004316	-3817 = 499 x 9 = 449	-3
START=2152 STOP=2166 INT=0000004474	-4104 = 370 x 9 = 333	-4
START=2167 STOP=2178 INT=0000002804	-2540 = 264 x 9 = 238	-3
START=2179 STOP=2203 INT=0000006932	-5188 = 1744 x 9 = 1570	
START=2204 STOP=2224 INT=0000002655	-2581 = 74 x 9 = 67	
START=2225 STOP=2248 INT=0000002062	-2041 = 21 x 9 = 19	

*10 KEV CH.*

MDA (nCi)
AM 241 .13
PU 238 7
PU 239 17
CM 244 8

**Chest results**

**Whole body results**

*K = 159  
C5137 = .3*

Figure 5-7. Example of alternate early in vivo bioassay form, with both WB and chest count results, March 29, 1977 [ORAUT 2023].



SAVANNAH RIVER PLANT IN VIVO COUNT RESULTS								
NAME	[REDACTED]	DEPT	SEP	HEIGHT (M)	1.70			
PRN	[REDACTED]	LOCATION	221-H	WEIGHT (KG)	75			
DATE	9 1 83	AGE	[REDACTED]	CHEST WALL (CM)	2.8			
TIME	10:59	SEX	[REDACTED]	COUNT, SEC	1800			
ROUTINE COUNT, REF DPSTN 2011 FOR INTERPRETATION.								
WHOLE BODY COUNT (NAI)				MDA@95%CL				
NUCLIDE	ENERGY(MEV)	GROSS	BKGD	NET	CALC	DIFF	COUNTS	NCI
CE-144	0.104 TO 0.145	7160	6259	901	1167	-266	403	16
CR-51	0.290 TO 0.349	7927	7852	75	621	-546	431	13
I-131	0.334 TO 0.400	6837	6665	172	521	-349	397	19
RU-106	0.449 TO 0.539	6508	6162	346	470	-124	301	10
CS-137	0.625 TO 0.704	4394	3914	480				
ZR-NB-95	0.710 TO 0.800	3612	3395	217	232	-15	282	8
ZN-65	1.050 TO 1.160	2774	2537	237	278	-41	249	4
CO-60	1.105 TO 1.250	3212	2932	280	314	-34	267	2
CO-60	1.250 TO 1.375	2199	2031	168	290	-130	227	2
K-40	1.344 TO 1.534	4617	3454	1163				
GM POTASSIUM = 116		NCI CS-137 = 1.69						
CHEST COUNT (NAI)								
EMISSION	ENERGY(MEV)	GROSS	BKGD	NET	CALC	DIFF	DL	
L X RAY	0.014 TO 0.022	659	506	153	156	-3	59	
	0.022 TO 0.047	1278	364	914	961	-47	84	
AM 241	0.040 TO 0.066	1242	369	873	922	-49	83	
	0.066 TO 0.089	1655	482	1173	1137	36	93	
K SCATTER	0.090 TO 0.127	1837	502	1335				
ISOTOPE	MDA @95% CL (NCI)							
AM 241	0.20							
PU 239	65							
PU 238	26							
CM 244	32							

Figure 5-8. Example of in vivo count report, September 1, 1983 [ORAUT 2023].

CANBERRA/RMC ABACOS-II		Termination Count Report			printed: 18-DEC-86 19:57:11	
name last: [REDACTED]		first: [REDACTED]		middle:		
social security / ID: [REDACTED]		date/time of exposure:		file number: 4157. input: 1		
counted: 18-DEC-86 19:51:42		operator: [300,001] BCW		FASTSCAN Diagnostic Counter No. 1 of 3		
employer: TRIT		job code/function: [REDACTED]		frequency of count:		
height: 72 ins.		weight (lbs.): 185		age: [REDACTED]		
sex: [REDACTED]		showered before count: N		wore special clothing for count: N		
comments:						
Final Whole Body Count Results (Nuclide Analysis)						
nuclide	activity (nCi)	% error (2.0 SD)	burden (%)	critical organ	comments	
K-40	126.2	15.6	0.0	TBODY		
CEPR-144	77.55				MDA activity used	
I-131	6.627				MDA activity used	
RURH-106	22.54				MDA activity used	
CS-137	4.873				MDA activity used	
ZR-95	9.177				MDA activity used	
NB-95	4.103				MDA activity used	
CO-60	4.051				MDA activity used	
<b>totals</b>	<b>126.2</b>		<b>0.</b>			
total % burden lung:		0.0	G.I.:	0.0	total body:	0.0
					thyroid:	0.0
					other:	0.0
analysis energy range: 89.5 to 2011.9 keV. nuclide library used: 100. last updated: 11-NOV-86						
calibration energy: 3 10-JAN-86 FWHM: 3 10-JAN-86 efficiency: 2 10-JAN-86 background file: 4027. 10-DEC-86						
Reviewed by: [REDACTED]		Date: 12/29/86				
Facility: [REDACTED]		*** E I DUPONT, SAVANNAH RIVER PLANT ***				

Figure 5-9. Example of in vivo count report, December 18, 1986 [ORAUT 2023].

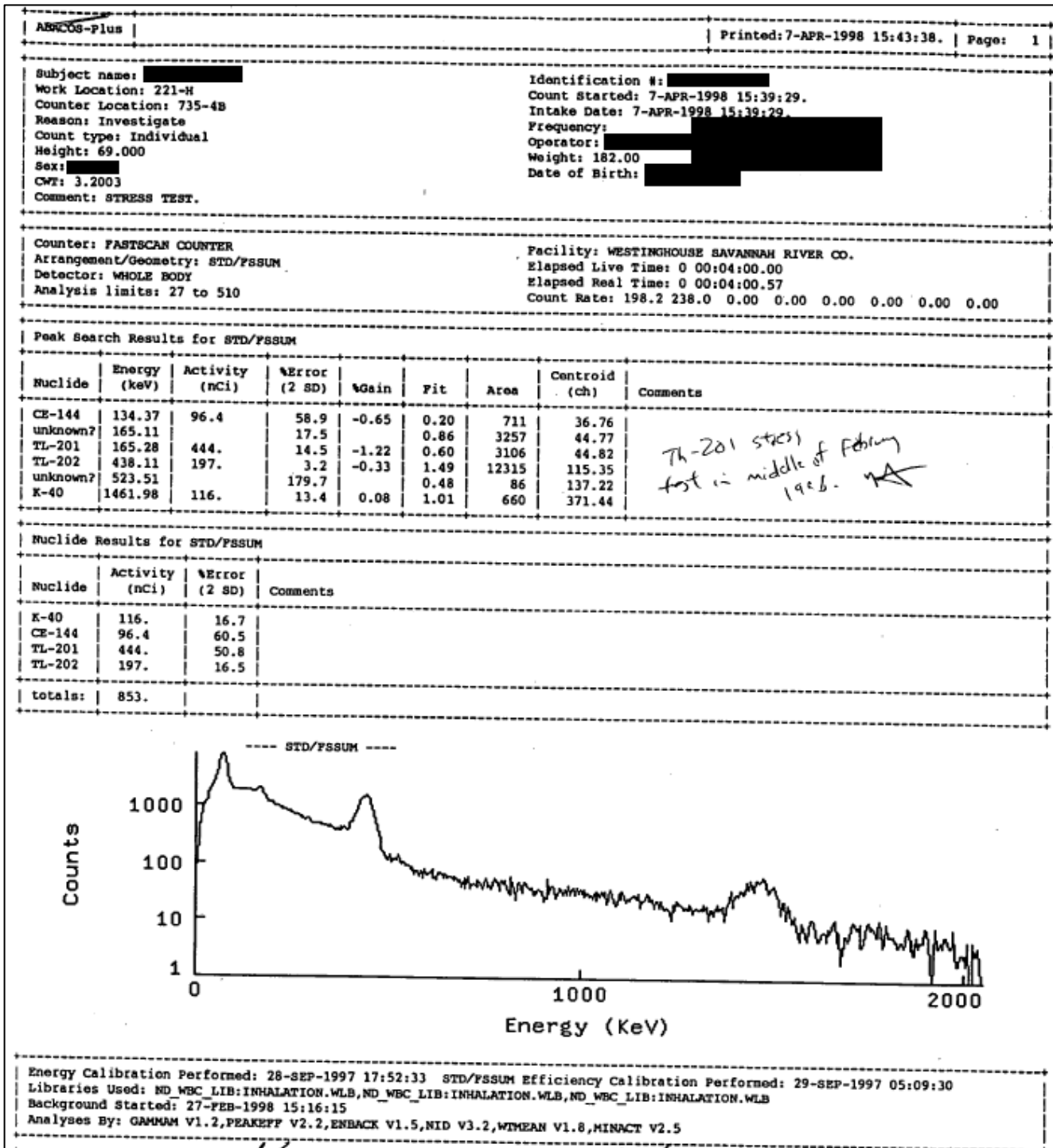


Figure 5-10. Example of WB count report used starting in 1995, April 7, 1998 [ORAUT 2023].

The FASTSCAN counters and the NaI(Tl) detector in the scanning bed counter were calibrated twice each year using the FASTSCAN Mixed Gamma Calibration Source. A Canberra FASTSCAN system was used for WB counting; these MDAs are valid for 1988 to 1991. SRS used the same MDA equation in 1992 through 2017. MDAs reported for 1997 were consistent with the MDAs reported for 1992; however, no MDAs were reported for <sup>58</sup>Co and <sup>59</sup>Fe. SRS did not include MDAs for some radionuclides in the 1997 version of the Internal Dosimetry Technical Basis Manual, and the 2001 version did not include some additional radionuclides. WB count MDA values determined for 1988 to the present are in Table 5-17. The MDA can be determined from the reported nuclide error

(Section 5.7.2). If the error is not reported, the default MDA can be used. Counting times to determine MDA are not provided in SRS documentation, but the counting for a routine WB count was 2 minutes starting with the use of the Canberra FASTSCAN [WSRC 1992, p. 15].

Table 5-17. MDAs for WB counting (nCi), 1988 to present.<sup>a,b</sup>

Nuclide	1988–1991	1992–1996	1997–2017	2018 <sup>c</sup>	2019–present
Np-237	13	14	14	14	18.3
U-235	13	14	14	14	Not applicable
Eu-152	16	18	18	24.5	Not applicable
Eu-154	10	8.4	8.4	9.15	Not applicable
Ce-141	14	16	Not applicable	Not applicable	Not applicable
Ce-144	55	69	69	69	239
Ba-140	17	16	Not applicable	Not applicable	Not applicable
La-140	3	2.7	Not applicable	Not applicable	Not applicable
Cs-134	4	3.8	3.8	3.8	4.6
Cs-137	3	4.1	4.1	4.1	4.8
I-131	4	5.4	Not applicable	Not applicable	Not applicable
I-133	5	4.6	Not applicable	Not applicable	Not applicable
Sb-125	13	14	14	14	Not applicable
Ru-103	5	4.8	Not applicable	Not applicable	Not applicable
Ru-106	25	36	36	43.8	Not applicable
Zr-95	5	6.2	Not applicable	Not applicable	Not applicable
Nb-95	2	3.4	Not applicable	Not applicable	Not applicable
Zn-65	4	6.1	6.1	6.1	7.0
Co-58	4	3.5	Not applicable	Not applicable	Not applicable
Co-60	4	2.9	2.9	2.9	3.1
Fe-59	6	5.6	Not applicable	Not applicable	Not applicable
Mn-54	4	3.4	3.4	3.7	Not applicable
Cd-109	Not applicable	Not applicable	Not applicable	Not applicable	1,632

a. Sources: WSRC [1992, p. 33; 1993a, p. 48; 1994, p. 55; 1996, p. 62; 1997, p. 59; 2001, p. 58]; SRNS [2008, p. 66; 2012, p. 75; 2014, p. 74]; Negron [2018, p. 8]; SRNS [2019, p. 46; 2020, p. 44].

b. Counting times are 2 min.

c. Although included in the 2018 memorandum *Minimum Detectable Activities of the In Vivo Counting Instrumentation* [Negron 2018], this MDA was not included in the 2019 or 2020 *Internal Dosimetry Technical Basis Manual* [SRNS 2019, 2020].

For special investigational counts using WB counting, the MDAs are stated to be approximately 1.414 lower than the MDA for the routine count [WSRC 2001, pp. 43–44].

Fallout affected everyone in North America, and body burdens of <sup>137</sup>Cs measurable in WB counters were common in the 1960s and 1970s. ORAUT-OTIB-0060 [ORAUT 2018] Section 2.4.4 contains guidance for evaluating nonoccupational cesium intakes.

#### 5.4.2 Chest Counting

SRS implemented a chest counter in the late 1960s using phoswich detectors. Initially the chest counter used a chair geometry, but SRS changed to a supine geometry in the 1970s [Taylor et al. 1995]. A phoswich detector is a combination of scintillators with dissimilar pulse shape characteristics optically coupled to each other and to a common photomultiplier tube. Pulse shape analysis distinguishes the signals from the two scintillators, identifying in which scintillator the event occurred. SRS incorporated six detectors: two phoswich on the chest used to detect TRU radionuclides and four 5-in.-diameter by 4-in. thick NaI(Tl) under the bed for fission and activation products [WSRC 1990, pp. 410–411]. MDAs were listed on results forms that varied from count to count but were in the neighborhood of 0.2 nCi for <sup>241</sup>Am, 5 nCi for <sup>238</sup>Pu, 10 nCi for <sup>239</sup>Pu, and 5 nCi for <sup>244</sup>Cm. Phoswich systems had better low-energy background sensitivity and better detection efficiency than germanium

systems. However, phoswich systems had poor resolution and could not discriminate between  $^{238}\text{Pu}$ ,  $^{239}\text{Pu}$ , and  $^{244}\text{Cm}$ . In 1976, SRS added side-looking germanium detectors to measure higher energy radionuclides deposited in the lungs. The bed counter was initially calibrated using a Remcal phantom. SRS used a torso skeleton with Remcal lungs for calibration, and in 1981 they changed to the use of a humanoid torso phantom developed by Lawrence Livermore National Laboratory. In 1989, SRS replaced the phoswich system with a germanium detector system using a reclining chair.

The earliest formats used to report chest count results from the onset of chest counting to late 1970s are shown in Figures 5-6 and 5-7.

Starting in September 1995 SRS used the same bed counter in Shielded Room #1, Building 735-4B described in the section above; the germanium detectors of the bed were used in chest counting [SRNS 2008]. SRS installed an additional commercial chest counter in a chair geometry, also with four germanium detectors, in Shielded Room #2. Both the bed and chair systems are still in use. SRS calibrates both counters using the same sources. Reported MDAs apply to both counters.

The Measurement and Test Equipment Department maintains the chest counters, which are calibrated at least annually. The system performance is at a minimum verified in control by use of the control charts and QC process any time new components are added or major changes are made. If the system operates within the control limits, calibration is performed at the next scheduled annual interval. If discrepancies are noted, calibration is performed before counting personnel. (The records the site provides should be used for dose reconstruction even if annual calibration was missed.)

The modern germanium detector system uses the software packages Abacos Plus and Abacos 2000 developed by Canberra Industries, Inc. The software calibration method establishes an energy calibration as a function of the multichannel analyzer channel location, an FWHM calibration as a function of energy, and a family of efficiency calibrations as functions of energy and chest wall thickness (CWT). The calibration method uses a mixed radiation source with several photon energies to determine the calibration parameters over the complete energy range. The lungs of the Humanoid phantom, the primary source for the chest counter, are similar in size and density to human lungs. The pairs of active lungs each contain known amounts of the radioisotopes  $^{238}\text{Pu}$ ,  $^{239}\text{Pu}$ ,  $^{241}\text{Am}$ ,  $^{238}\text{U}$ ,  $^{235}\text{U}$ ,  $^{241}\text{Am}$ ,  $^{152}\text{Eu}$ , or  $^{237}\text{Np}$  [WSRC 1990, pp. 436–437; 1992, pp. 70–80; 2001, pp. 195–202]. Additional lung sets for efficiency calibrations for such as  $^{235}\text{U}$  and  $^{238}\text{U}$  can be added during calibration or during an incident. Efficiency responses are directly measured only for the five CWTs of the phantom with no overlay and the phantom with each of the four overlays. These values give efficiency measurements for CWTs ranging from 16 to 40 mm. The Abacos Plus and Abacos 2000 software packages interpolate the information stored from these counts to calculate the efficiency for a particular CWT.

Note that, in the 1990s, SRS HP counted workers using a 2-minute scan with the FASTSCAN. Based on results of the 2-minute count, some workers were asked to change clothes and then HP performed a chest count. Most workers were counted in a chair counter with a germanium detector, but some were too large for the chair and were counted on a bed counter. The bed counter had both an NaI(Tl) detector and a GeLi detector configuration that worked in unison; the results of both counters were included on a count report. However, HP only considered results from the germanium configuration. The results from the NaI counter should not be considered. An example is shown in Figure 5-11 of a chest count on April 7, 1998. The bed NaI detector is denoted as STD/SBNAI. Results in that bottom section of the report should be ignored.

MDAs for later periods have been generated individually for counts by processing software, with typical values in Table 5-18, based on a CWT of 2.5 cm. However, MDAs are often listed for each count as part of the measurement results, and the count-specific MDAs should be used when available. The CWT is an important factor in quantifying the amount of an actinide in the chest. In

1970, ultrasound equipment was purchased to measure the chest-wall thickness. Through the years, the equipment was abandoned and upgraded several times.

Count type: Individual		Operator: [REDACTED]						
Height: 70.500		Weight: 196.50						
Sex: [REDACTED]		Date of Birth: [REDACTED]						
CMT: 3.4954								
Comment: ROUTINE CHEST COUNT/DOOR CLOSED								
Counter: BED/4 LUNG DETS		Facility: WESTINGHOUSE SAVANNAH RIVER CO.						
Arrangement/Geometry: STD/SBLSUM		Elapsed Live Time: 0 00:30:00.00						
Detector: LUNGS		Elapsed Real Time: 0 00:30:00.56						
Analysis limits: 50 to 2040		Count Rate: 1.717 1.498 1.547 1.642 0.00 0.00 0.00 0.00						
Counter: BED/NAI DETECTOR		Facility: WESTINGHOUSE SAVANNAH RIVER CO.						
Arrangement/Geometry: STD/SBNAI		Elapsed Live Time: 0 00:30:00.00						
Detector: WHOLE BODY		Elapsed Real Time: 0 00:30:01.36						
Analysis limits: 5 to 510		Count Rate: 59.24 0.00 0.00 0.00 0.00 0.00 0.00 0.00						
Peak Search Results								
----- No Peaks Found -----								
Nuclide Results for STD/SBLSUM								
Nuclide	Activity (nCi)	%Error (2 SD)	Comments					
CS-137	< 2.89		MDA activity reported					
PB-212	<0.213		MDA activity reported					
PB-214	<0.282		MDA activity reported					
AC-228	<0.840		MDA activity reported					
TH-234	< 1.55		MDA activity reported					
U-235	<0.124		MDA activity reported					
NP-237	<0.425		MDA activity reported					
PU-238	< 182.		MDA activity reported					
PU-239	< 472.		MDA activity reported					
AM-241	<0.172		MDA activity reported					
----- No Nuclides Identified -----								
totals: 0.000E+00								
Energy Calibration Performed: 4-FEB-1998 15:45:49. STD/SBLSUM Efficiency Calibration Performed: 6-FEB-1998 09:04:28.								
Libraries Used: ND_WBC_LIB:LOW-ENERGY.WLB,ND_WBC_LIB:LOW-ENERGY.WLB,ND_WBC_LIB:LOW-ENERGY.WLB								
Analyses By: GAMMAN V1.2,PEAKEFF V2.2,NID V3.2,WMEAN V1.8,MINACT V2.5								
Peak Search Results for STD/SBNAI								
Nuclide	Energy (keV)	Activity (nCi)	%Error (2 SD)	%Gain	Fit	Area	Centroid (ch)	Comments
unknown?	73.65		39.4		1.57	664	22.28	
unknown?	236.00		12.8		1.26	2173	64.34	
EU-152	346.21	14.5	16.1	0.53	0.97	1277	92.76	
unknown?	599.95		17.7		0.82	988	157.79	
unknown?	911.67		30.6		0.85	508	236.95	
EU-152	1120.13	11.8	35.7	0.77	0.81	372	289.44	
EU-152	1410.07	6.82	47.6	0.14	0.84	306	361.86	
K-40	1462.92	122.	5.5	0.19	0.90	2776	375.00	

Figure 5-11. Example of chest count report in a format starting in 1995, April 7, 1998 [ORAUT 2023].

### Plutonium/Uranium Lung Counts (starting 1989)

This guidance is provided to address the possibility of uranium exposure using the plutonium/uranium lung counter results. These counts could also be used to limit other nuclides, but uranium will most frequently be encountered.

Starting in December 1989, the site began using a six-detector, solid-state planar germanium array in a reclining chair geometry for the lung counter. In 1995, two larger detectors replaced the six-detector array and at the same time, a move was made to a new in vivo counting facility. Eventually, an empirical height-weight-to-CWT algorithm was adopted [WSRC 2001, pp. 197–198] to estimate the CWT to replace the use of ultrasound measurement. This algorithm, which is still in routine use today, is known to be inaccurate for females. Inaccuracies in the algorithm influence how much plutonium dose reconstructors calculate to be present, but do not influence the true ability to detect the plutonium. An example of a report commonly seen for the Plutonium/Uranium Lung Counter is shown in Figure 5-12. It has the plutonium/uranium counter identified as the location on reports. Note that there was only a single lung counter in use at a time, so in this period uranium is implied even if it is not explicitly stated in the counter location.

In 1990, SRS listed new MDA values for both the germanium detector counter and the phoswich counter. For the germanium counter MDA was defined using Equation 5-3. For the phoswich counter MDA was defined as [WSRC 1990, p. 434]:

$$MDA = (2.71 + 4.66 \times (background + gross)^{1/2}) \times conv. factor \quad (5-4)$$

where

- MDA* = minimum detectable activity (pCi/L)  
*background* = 30-min gross background count in energy region of interest  
*gross* = 30-min gross count in energy region of interest  
*conv. factor* = conversion factor of nCi per counts/count time (min)

Counts to determine chest count MDA were typically done for 30 minutes starting in 1990. The chest counters were calibrated using phantoms and standards described in WSRC [1990]. MDA values reported for chest counting valid for 1988 through 1991 are in Table 5-18.

Table 5-18. MDAs for chest counting (nCi), 1988 to 1991.<sup>a</sup>

Nuclide	Germanium <sup>b</sup>	Phoswich <sup>c</sup>
Am-241	0.13	0.16
Cf-252	30	None
Cm-242	27	23
Cm-244	29	25
Pu-238	43	20
Pu-239	110	40
Pu-240	46	22
Np-237	0.35	0.48
U-234	43	None
U-235	0.1	None
U-236	91	None
U-238	1.1	2
Th-228	3.2	None
Th-232	28	None
Eu-152	0.13	None
Ce-141	0.1	None
Ce-144	0.43	None

- a. Counting times are not documented.  
b. Source: WSRC [1990, p. 199].  
c. Source: WSRC [1990, pp. 161–166, 168, 170, and 172].





Nuclide	Period	MDA
Cf-252	1992–2017	32
Cf-252	2018 only	42
Cf-252	2019–present	36
Cm-242	1992–2011	28
Cm-244	1992–2017	37
Cm-244	2018 only	49.5
Cm-244	2019–present	40
Pu-238	1992–2017	58
Pu-238	2018 only	66
Pu-238	2019–present	55
Pu-239	1992–2017	130
Pu-239	2018 only	174.4
Pu-239	2019–present	143
Pu-240	1992–2017	47
Pu-240	2018 only	69
Pu-240	2019–present	57
Np-237	1992–2017	0.31
Np-237	2018 only	0.46
Np-237	2019–present	0.4
U-234	1992–2017	30
U-234	2018 only	61.9
U-234	2019–present	60
U-235	1992–2017	0.1
U-235	2018 only	0.13
U-235	2019–present	0.1
U-236	1992–2017	89
U-236	2018 only	123
U-236	2019–present	107
U-238	1992–2017	1.1
U-238	2018–present	1.7
Th-228	1992–2017	3.2
Th-228	2019–present <sup>c</sup>	Not evaluated
Th-232	1992–2017	31
Th-232	2018 only	34.9
Th-232	2019–present	34
Eu-152	1992–2017	0.056
Eu-152	2018 only	0.17
Ce-141	1992–1996	0.071
Ce-144	1992–2017	0.31
Ce-144	2018 only	0.55
Ce-144	2019–present	0.5

- a. Sources: WSRC [1992, p. 33; 1993a, p. 48; 1994, p. 55; 1996, p. 62; 1997, p. 59; 2001, p. 58]; SRNS [2008, p. 66; 2012, p. 75; 2014, p. 74]; Negron [2018, p. 8]; SRNS [2019, p. 46; 2020, p. 44].
- b. Counting times are 30 min.
- c. Thorium-228 is measured in vivo by chest counting based on the 239-keV photon for <sup>212</sup>Pb (assuming <sup>212</sup>Pb is in secular equilibrium with <sup>228</sup>Th) or the 84.4-keV photon of <sup>228</sup>Th and 2.8-cm CWT. Chest count MDAs have not been evaluated for <sup>228</sup>Th [SRNS 2019, pp. 43–44].

MDAs are affected by uncertainties in the system of measurement. The uncertainty in the MDA calculation has been estimated, according to WSRC-IM-90-139 Revision 8, as  $\pm 46$  wt% [WSRC 2001]. The uncertainty in CWT can have a large impact on MDA determination. The estimations of uncertainties as published in 2001 are in Table 5-20.

Table 5-20. Estimates of uncertainty for chest counting MDAs.<sup>a</sup>

Type	Uncertainty (%)
Variation in chest depth	±12
Variation in CWT	±15
Variation in activity location	±5
Variation in detector placement	±5
Variation in subject background	±10
Calibration	±5
Counting statistics	±40
<b>Total propagated uncertainty</b>	<b>+46</b>

a. Source: [WSRC 2001, p. 202].

For special investigational chest counts, the MDAs are stated to be approximately 0.77 times the uncertainty values in Table 5-20 [WSRC 2001, p. 44].

## 5.5 PERSONAL AIR SAMPLING DATA

Starting in about 2004, SRS used PAS because it was not practical to use regional air samplers (for work outside of buildings) inside plastic suits used in highly contaminated environments and for operational studies.

Air sample records were considered workplace monitoring records as opposed to intake monitoring records and therefore were not maintained in the same records collection as bioassay records. Air sample records were associated with each facility separately and were stored as facility records. Boxes containing workplace monitoring records can be retrieved from storage at the federal repository, but locating a specific set of air sample records would be very labor consuming and should be considered only as a last resort.

Dose assigned from PAS data might be present in the annual dose summary report. Dose assigned based on PAS data is not directly applicable to the organ of interest when performing a dose reconstruction. If a committed effective dose is recorded and no bioassay data were provided for the dose assignment, the dose reconstructor should make a request for additional data. If the site provides information that the dose was based on PAS (no bioassay data is available for the assignment of dose), contact the Principal Scientist for Internal Dosimetry for guidance.

## 5.6 ASSESSMENT OF INTAKE

General instructions of assessment of intakes are provided in the following to support guidance provided in ORAUT-OTIB-0060 [ORAUT 2018].

### 5.6.1 Trivalent Actinides (Americium, Curium, and Californium)

Use of a gross alpha count before 1994 means that the measurement itself does not indicate which radionuclide or combination of radionuclides was measured. Location or other information about an intake might allow determination of the principal radionuclide. If possible, based on work history, the dose reconstructor should assign intakes and doses for the radionuclide and absorption type that is most favorable to the claimant for workers in the locations and for the time periods listed in Table 5-21. Fresh <sup>241</sup>Am was only a concern in certain areas of Building 773-A and the Multi-Purpose Processing Facility.

Table 5-21. HP Area codes for locations where <sup>241</sup>Am, <sup>244</sup>Cm, and <sup>252</sup>Cf were used.<sup>a</sup>

Period	221-H Area Outside Facilities <sup>b</sup>	735-A and 735-11A	E-Area Solid Waste Disposal Facility (643-G Burial Grounds, 618-G Class, Yard)	773-A, 776-A, and 723-A	Not identifiable or unknown <sup>c</sup>	Multi-Purpose Processing Facility in 221-F (Am only)	Waste Certification Facility (724-8E and WCF Building start 12/1986)	Central Shops (includes CTW-specific HPA Area codes)
1959–1972	2A	6F	12A	5A, 6N	None	1A	None	1Z, 12D, 12E, 12Z
1973–08/16/1990	1H, 2H	5D	12B, 4F, 3G, 8G	5A, 7L	7Y, 8D, 8E, O00, missing (cannot find on HPA code list)	1F, 2F	None	1A, 5W, 7A, 7B, 7J, 7R, 7T, 8A, 8B, 8C, 8H, 8I, 8M, 8P, 8S, 8T, 8K, 8L, 7Y
08/17/1990–2003	H01, H02	5A (773-A)	B12, G03, F04, G08	A15, A27	J25, J26, Y01, missing (cannot find on HPA code list)	F01, F02	None	A18, F01, J01, J02, J03, J05, J06, J08, J10, J12, J14, J17, J18, J20, J21, J24, J30, J31, Y01
2004–present	HCA	SRT (apply 773-A intakes)	SWM	SRT, SRE (apply 773-A intakes)	Blank, any code not already listed	FCA (F Canyon)	SSS	Use occupation given in the claims tracking application

a. Source: ORAUT [2014a].

b. Note that HP Area Codes alone are not sufficient to isolate a worker to certain individual locations such as A-Line, and without other evidence radionuclides given for H Canyon should be assumed.

c. Unknown facility radionuclides should only be assigned if no information is available from any source about the worker's work location.

Internal trivalent co-exposure intake rates from ORAUT-TKBS-0003-7 may be used for dose assignment when monitoring data are lacking (incomplete or unavailable) [ORAUT 2024b, Table 7-18].

### 5.6.2 Plutonium

Plutonium was created within fuel and target assemblies in the reactors through neutron irradiation of uranium. It was separated in F-Canyon and in H-Canyon and then shipped to other facilities or manufactured on site into new target assemblies for the production of heavier elements, termed transplutonium elements.

The isotopic ratios in the plutonium varied with the isotopic mix of the uranium targets and the length of neutron irradiation. Irradiation of  $^{238}\text{U}$  produces  $^{239}\text{Pu}$ ,  $^{240}\text{Pu}$ , and lesser amounts of heavier isotopes. Plutonium produced for weapons was greater than 90%  $^{239}\text{Pu}$  and about 6% to 8%  $^{240}\text{Pu}$ . This was referred to as “weapons-grade” or “low exposure” plutonium [DOE 2006, p. 18]. In weapons grade plutonium, about 0.5% is  $^{241}\text{Pu}$ . This can be increased to 15% to 20% in “high burnup” plutonium [Healy 1977, p. 86]. Very long irradiation of plutonium increased the fraction of  $^{242}\text{Pu}$  and decreased the fractions of other isotopes. If plutonium assaying at 52%  $^{242}\text{Pu}$  was irradiated for another 2.5 years, it could reach an estimated 97.5%  $^{242}\text{Pu}$  [DuPont 1974b, p. 26]. Isotopic ratios for a high  $^{242}\text{Pu}$  mixture from the Curium I campaign are listed in Table 5-5 and should be used to assign intakes to workers in F Area for 1963 to 1967.

Plutonium was also used as target material to produce heavier elements. Prolonged irradiation of  $^{239}\text{Pu}$  produces  $^{242}\text{Pu}$ ,  $^{243}\text{Am}$ , and  $^{244}\text{Cm}$ . In the first transplutonium campaign at SRS,  $^{239}\text{Pu}$  slugs manufactured in Hanford were loaded into tube assemblies at SRS and then irradiated in the reactors starting in June 1959 [DuPont 1971c, pp. 17–20].

Plutonium urinalysis results and fecal results before 1981 (and before 1988 for special samples) measured the total activity from alpha-emitting isotopes. Since  $^{238}\text{Pu}$  and  $^{239+240}\text{Pu}$  results were reported separately from 1981 to the present, the two sets of data cannot be entered into the intake assessment code as reported. If, for a specific case, all the bioassay data are before 1981, the total plutonium alpha data can be entered as  $^{239}\text{Pu}$ . The calculated intake will have to be partitioned into  $^{238}\text{Pu}$  and  $^{239}\text{Pu}$  per the mixture ratios applicable to the intake, if known. However, it is unlikely that for the older intakes, intake-specific isotopic mixture ratios were established, so the default mixture ratios should be used. The same mixture ratio will then be used to determine the intake of  $^{241}\text{Pu}$  and  $^{241}\text{Am}$ . Note that in the chemistry techniques described above, the gross alpha count would not have included any  $^{241}\text{Am}$  present in the bioassay sample.

If, for a given intake, the plutonium mixture composition can be determined from the source of the contamination, that activity composition should be used. If unknown, the default ratios provided in Table 5-4 in Section 5.3.2.2 can be used as best fits the nature of the intake. Consult ORAUT-OTIB-0060 for guidance on mixture selection [ORAUT 2018, Section 3.7.1.2.1].

If some of the bioassay data are results reported before 1981 and some after, those data must be made consistent in one of two ways. First, if all the  $^{238}\text{Pu}$  and  $^{239}\text{Pu}$  results are above their respective MDAs, then the  $^{238}\text{Pu}$  and  $^{239}\text{Pu}$  results for each sample can be summed to produce a total plutonium alpha result. This result can be equated to the pre-1981 results and treated in the manner described in the preceding paragraph. Second, if some of the  $^{238}\text{Pu}$  or  $^{239}\text{Pu}$  results are below their MDAs, then it could be best to use the isotope that has the most robust dataset and adjust each of the pre-1981 results to reflect just the portion from that chosen isotope. The adjustment should be made using the default isotopic mixture ratios unless the intake-specific ratio is known. For example, a gross alpha result of 2.1 dpm/1.5 L would be converted to 1.9 dpm/1.5 L of  $^{239}\text{Pu}$  for 6 wt% Pu aged 10 years or 1.8 dpm/1.5 L for 12 wt% Pu aged 10 years before being entered into the intake code.

Plutonium-239 mixtures at SRS could have existed as absorption types M, S, or SS [ORAUT 2020b].

Dose reconstructors should model the <sup>241</sup>Am that is a component of plutonium contamination in the lung the same as the plutonium matrix in which it has grown. In other words, the americium should be treated as absorption type S if the plutonium is type S. Refer to ORAUT-OTIB-0060 Section 3.7.1.3, “Americium Ingrowth to Plutonium Mixture,” for guidance on using <sup>241</sup>Am results for the assessment of plutonium mixtures [ORAUT 2018]. Note that this does not apply to separated <sup>241</sup>Am.

Plutonium co-exposure intake rates as outlined in ORAUT-TKBS-0003-7 and ORAUT-OTIB-0060 may be used for dose assignment when monitoring data are lacking (incomplete or unavailable) [ORAUT 2024b, 2018] and may be extended past 1990. Consult ORAUT-OTIB-0060 for guidance.

**5.6.3 Plutonium-238**

DCAS-RPT-005, *Alternative Dissolution Models for Insoluble Pu-238* [NIOSH 2018], should be used to apply a third dissolution type to <sup>238</sup>Pu assessments for insoluble material characterized as nonmonotonic (NM) in addition to the standard Types M and S for work from 1963 through 1984 [ORAUT 2016]. The report assigns the Los Alamos model to SRS cases; this model is designated “NM” in the Web CAD tool. Only positive <sup>238</sup>Pu results should be evaluated; considering Type M, S, or NM dose. When overestimating, the positive <sup>238</sup>Pu dose is added in addition to the Pu mixture dose based on <sup>239</sup>Pu. If the dose reconstruction is a best estimate, contact the Principal Scientist for Internal Dosimetry for guidance. This model should only be applied to workers who worked in Building 773-A, Building 235-F, old and new HB-Lines, and subcontractor CTWs for the periods in Table 5-22, and with one of those locations mentioned in the claimant computer-assisted telephone interview to determine those workers. The dissolution type that results in the highest dose to the organ of interest should be used for the final dose assignment. DCAS-RPRT-005 also provides instructions for the dose reconstructor to use the insoluble dissolution type in the Integrated Modules for Bioassay Analysis (IMBA).

Table 5-22. HP Area codes for insoluble <sup>238</sup>Pu.<sup>a</sup>

Period	773-A, 776-A, and 723-A	235-F Vaults	Old HB-Line (1963–1983) New HB-Line (1985–2011)	Central Shops (Includes CTW specific HPA Area codes)
1959–1972	5A, 6N	1A, 1F	2A	1Z, 12D, 12E, 12Z
1973– 08/16/1990	5A, 7L	1F, 2F, 8F	1H, 2H	1A, 5F, 5W, 5H, 6H, 7A, 7B, 7J, 7R, 7T, 8A, 8B, 8C, 8H, 8I, 8M, 8P, 8S, 8T, 8K, 8L, 7Y
08/17/1990– 2003	A15, A27	F01, F02, F08	H01, H02	A18, F05, H05, H06, J01, J02, J03, J05, J06, J08, J10, J12, J14, J17, J18, J20, J21, J24, J30, J31, Y01
2004–present	SRT, SRE	235	HBL (ended in 2011)	Use occupation given in the computer-assisted telephone interview and/or DOL records

a. Source: ORAUT [2014a].

**5.6.4 Tritium**

Assess tritium dose (HTO only) for workers with data using the information in Section 5.3.2.3 of this document and the guidance in ORAUT-OTIB-0011, *Technical Information Bulletin: Tritium Calculated and Missed Dose Estimates* [ORAUT 2004a].

### 5.6.4.1 Unmonitored Tritium Dose

Tritium dose values in ORAUT-TKBS-0003-7 [ORAUT 2024b] should be used for workers who were unmonitored for tritium exposure for 90 days or greater while assigned to known tritium areas including all SRS reactors, 221-H (H Canyon) outside areas and unknown areas, Building 232 (New Manufacturing Facility), H Area Old Manufacturing Facility, Tritium Complex, Defense Waste Processing Facility, D-Area (400-D), and Z-Area. Consult ORAUT-TKBS-0003-2 for more information on those facilities [ORAUT 2024a].

### 5.6.4.2 Special Tritium Compounds

In the early 1980s, SRL began a major program to develop and use metal hydrides in tritium production. Metal tritides form hydrides that absorb and store tritium atoms in the crystalline structure of the metal. These metals include uranium, palladium, titanium, lanthanum-aluminum-nickel alloy (LANA.75), mischmetal, and zirconium. Evaluations of the physical and chemical properties of these tritium compounds have shown that uranium and palladium do not bind tritium in a manner significantly different from HTO. The remaining metals form compounds that cannot be adequately assessed using the HTO urine bioassay model. A study by Lovelace Respiratory Research Institute in 2011 concluded that in vitro dissolution of LANA.75 occurs very rapidly with less than 1% of the original tritium activity remaining after 10 days [Farfán et al. 2011; ORAUT 2017a]. This dissolution was assigned as Type F. However, SRS obtained additional information in 2012 that raised awareness about the possible introduction of a new species of physiologically uncharacterized stable metal tritides (SMTs).

The solubility types of representative SMTs in Table 5-23 should be used when the material is known. When unknown, possible material Types (F and M) should be assessed and the highest dose assigned in accordance with ORAUT-OTIB-0066, *Calculation of Dose from Intakes of Special Tritium Compounds* [ORAUT 2020c].

Table 5-23. Representative SMT absorption types.<sup>a</sup>

SMT	Absorption type
Lana.75	F
Palladium	F
Palladium-aluminum	M
Palladium-silver	F
Uranium	F
Titanium	M
Zirconium	M
Zirconium-iron-chromium	M

a. Source: ORAUT [2017a, p. 26].

Tritium was stored as metal tritides starting in the mid-1970s in the tritium facilities 232-H, 233-H, 234-H, and 238-H. These sources would have been minor sources of intake relative to the total activity as HTO or elemental tritium. However, intake of metal tritides cannot be completely ruled out and bioassay sampling for particulate forms of tritium was not performed. The surface contamination limit for tritium in accessible spaces was  $10^6$  dpm/100 cm<sup>2</sup>. While most contamination would have been HTO, conservatively assuming 100% was metal tritides and applying a resuspension factor of  $5 \times 10^{-5}/m$  [ORAUT 2013c], which would apply to aggressive disturbance of the surfaces, then:

$$\begin{aligned} \text{air concentration}_{\text{mt}} &= 1 \times 10^6 \text{ dpm}/100 \text{ cm}^2 \times 1.0 \times 1 \times 10^4 \text{ cm}^2/\text{m}^2 \times 5 \times 10^{-5}/\text{m} \\ &= 5,000 \text{ dpm}/\text{m}^3 = 2,252 \text{ pCi}/\text{m}^3 \end{aligned} \quad (5-5)$$

The daily intake is determined by the air concentration times the breathing rate per year divided by 365 d/yr:

$$2,252 \text{ pCi/m}^3 \times 2,400 \text{ m}^3/\text{yr}/365 \text{ d/yr} = 14,800 \text{ pCi/d} \quad (5-6)$$

Tritium from this assumed SMT intake would result in expected excretion rates that would likely be undetectable; therefore, no adjustment of urine sample results is needed for the HTO/OBT intake assessment.

For a chronic 1-year intake, only 5 organs have committed equivalent doses greater than 1 mrem for Type F or Type M solubilities. These are:

- Lower large intestine, LLI = 2.23E-03 rem
- Lung = 2.98E-03 rem
- Colon = 1.41E-03 rem
- Alveolar-interstitium, AI = 8.88E-03 rem
- Thoracic lymph nodes, LN(TH) = 1.44E-03 rem

Given there is no Type S material at SRS, almost all the committed dose occurs in the first year of exposure. Therefore, for these organs, the SMT dose can be assessed by assigning 100% of the committed equivalent doses listed above in the year of the intake. For all other organs, the dose associated with SMT is considered insignificant and need not be assessed.

Exposure to organically bound tritium (OBT) in the same tritium facilities was possible. Milham and Boni [1976] describe measurements of tritium species in effluents from the tritium processing areas, primarily tritium methane, but the measured OBTs were less than 1% of the total tritium release. Howard [2000] and Rabun [2000] describe potential exposure to OBT exposure from contaminated mercury diffusion pumps and mechanical vacuum pumps in the Material Test Facility in Building 232-H as early as 1957 and from work in hoods in Building 773-A. OBT compounds were manufactured in the Material Test Facility. Only incidental quantities would have been encountered in 773-A.

With most exposures to tritium, it is not possible to identify the tritium compound taken into the body based on the observed urinary excretion. Therefore, the selection of the appropriate tritium compound in an intake evaluation must be based on process knowledge of the source terms in the workplace. Dose from SMTs should be considered for work in Building 773-A and tritium processing areas in H Area and if the worker was monitored for tritium. Consult the guidance in ORAUT-RPRT-0072, *Locations of Stable Metal Tritide Use at the Savannah River Site* [ORAUT 2017a]. Exposures to SMTs were limited to the metals listed in Table 5-23 above and certain locations in A-, F-, and H-Areas. SMT doses using the intake rate from Equation 5-6 should be assigned to any worker with tritium monitoring data working in F- and H-Areas and Building 773-A in using the HP badge locations and periods provided in Table 5-24, in addition to doses received from tritium or OBT.

Table 5-24. HP Area codes for SMT/OBT dose assignment.<sup>a</sup>

Period	773-A	H-Area	Central Shops (includes CTW-specific HPA Area codes)
1973–08/16/1990	5A	2H, 4H,	1A, 5F, 5W, 5H, 6H, 7A, 7B, 7J, 7R, 7T, 8A, 8B, 8C, 8H, 8I, 8M, 8P, 8S, 8T, 8K, 8L, 7Y
08/17/1990–2003	A15	H01, H02, H03, H04	A18, F05, H05, H06, J01, J02, J03, J05, J06, J08, J10, J12, J14, J17, J18, J20, J21, J24, J30, J31, Y01
2004–present	SRE, SRT	HCA, TEF (tritium extraction facility), TRI (DP tritium facilities)	Use occupation given in the computer-assisted telephone interview and/or DOL records

a. Source: ORAUT [2014a].

Doses from OBT should be assigned to any worker in the locations in Table 5-24 who also has tritium bioassay data using those bioassay results and only for periods where a worker has tritium bioassay data. In addition to doses from the SMT intakes derived above, tritium bioassay results in the given timeframes should be assessed as OBT and HTO and the larger of those two doses assigned.

### 5.6.5 Uranium

Initially, plutonium was produced from uranium within the same fuel elements that operated the reactor. Starting in the mid-1950s, the reactors began using separate assemblies of NU fuel and DU targets. All SRS reactors used NU as fuel until 1968 when the reactors were modified to use EU [SRNS 2013b, p. 185].

NU and DU were primarily supplied by the Feed Materials Production Center (FMPC). EU was primarily supplied by Y-12. Significantly more DU than EU was used. During operations, approximately 185 metric tons of EU was sent by Y-12 and 46,206 metric tons of DU was sent by FMPC. Detailed information on uranium material flow is found in the *Historical Generation and Flow of Recycled Uranium at the Savannah River Site* [McCarty 2000]. There were two general uranium cycles at SRS.

EU was:

- Shipped from Y-12,
- Manufactured into fuel elements in the 300-M area,
- Used in the reactors,
- Separated from plutonium and FPs in H Canyon, and
- Sent in solution to Y-12 to be made into metal for reuse.

DU was:

- Shipped from the FMPC,
- Manufactured into target elements in the 300-M area,



- Irradiated in reactors,
- Separated from the plutonium and FPs in F Canyon, and
- Sent in solution to the FMPC to be made into metal for reuse.

The HP area location codes Table 5-25 lists the areas, the types of uranium handled, and the periods of use. This table includes EU, DU, NU, and two types of uranium related to limited campaigns, <sup>233</sup>U, processed in H-Canyon from 1964 through 1969, and TRM, processed in H-Canyon from 2017 through 2020.

Table 5-25. Uranium areas and periods of use.

Location	Types and dates used
320-M, 321-M <sup>a</sup>	NU December 1953–1967 EU 1954–1988 DU 1955–1988
Reactors <sup>b</sup>	NU December 1953–1967 EU 1968–1988 DU 1955–1988
H Canyon–A-Line, HB Line (221-H), Outside Areas <sup>c</sup>	DU 1955–present EU May 1959–present U-233 1964–1969 TRM 2017–2020
F Canyon–A-Line, FB Line (221-F), Outside Areas <sup>d</sup>	NU November 1954–1967 DU 1955–2006
Central Shops / CTW <sup>e</sup>	DU 1955–present EU May 1959–present U-233 1964–1969 TRM 2017–2020
772-F, 772-1F <sup>f</sup>	DU 1955–2021 EU May 1959–2021 U-233 1964–1969 TRM 2017–2020
723-A, 773-A, 735-A, 735-11A, 776-A <sup>g</sup>	DU 1955–present EU May 1959–present U-233 1964–1969 TRM 2017–2020
777-M <sup>h</sup>	NU December 1953–1967 EU 1968–1988 DU 1955–1988
CMX [679-T] (09/01/1951–1983) and TNX [678-T] (12/10/1951–09/2004) <sup>i</sup>	NU December 1953–1967 EU 1968–1988 DU 1955–1988

- Received EU, DU, and NU from Y-12 and FMPC. NU shipments to SRS began in 1953, DU shipments to SRS began in 1955, and EU shipments to SRS began in 1954 [McCarty 2000, pp. 75, 84–85]; manufactured uranium into fuel elements and into target elements [WSRC 2006b, p. 42].
- Operated with NU fuel elements after startup, with EU fuel elements after 1968 [Phifer et al. 2008, p. 23]; irradiated DU targets to make plutonium [SRNS 2013b, p. 185].
- Separated plutonium from DU target material using plutonium-uranium extraction (PUREX) process [SRNS 2013b, pp 181–182, 185]; separated plutonium from EU after May 1959 using HM process [SRNS 2013b, p. 195]; separated U-233 between 1964 and 1969 [Till 2001, p. 102]; processed TRM between 2017 and 2020 [National Nuclear Security Administration (NNSA) 2021, p. 3].

- d. Starting in November 1954, separated plutonium from irradiated NU target material utilizing the PUREX process. Starting in 1968, separation of plutonium from irradiated DU target material [Severynse 1998, p. 7].
- e. CTWs might have worked anywhere on site. Ford Building at Central Shops used to repair contaminated reactor heat exchangers since at least 1961 and the mid-1980s [DuPont 1961, p. 282].
- f. The production control laboratories, 772-F and 772-1F, were near the 221-F Canyon Facility and provided analytical support for the 200-F and 200-H separations processes [SRNS 2013b, p. 151]. Thorium oxide irradiation to produce 130 kg of U-233 began in 1964 [DuPont 1984a, p. 94]. The Thorex IIB campaign in the plant was completed in 1969 with the recovery of 181 kg of U-233 [DuPont 1984b, p. 193]. 772-F and 772-1F ended operations in 2021 [ORAUT 2024a, Table 2-21].
- g. The A-Area (700 Area with building numbers in the 700s) also contains the Savannah River National Laboratory (SRNL; Building 773-A), Main Laboratory Buildings (735-A and 735-11A), and the Liquid Waste Handling Facility (Building 776-A). All are still operating [ORAUT 2024a, Tables 2-47, 2-48, and 2-49].
- h. Contained the Physics Laboratory with three test reactors: the Process Development Pile, the Standard Pile, and the Subcritical Experiment test reactors that were used to test the fuel and targets manufactured in the 300 Area [WSRC 2006b, pp. 37, 48, 113]. The reactors in 777-M were effectively shut down in the 1980s after most of their functions had been assumed by computer modeling [Strack 2002, p. 413].
- i. Process pilot plants that tested fuel and target elements. TNX construction began in May 1951 and was completed by the following fall. The first uranium had been introduced and pilot runs began December 10, 1951 [Strack 2002, p. 417]. TNX has continued to serve as a pilot plant for operations in the F and H Areas [Strack 2002, p. 418]. CMX construction began in April 1951, and the facility was turned over to operations in September of the same year [Strack 2002, p. 413]. After 1983, the testing of new fuel and target elements was moved from CMX to SRL [WSRC 2006b, p. 48].

Uranium was blended to other enrichments in the 300-M area [WSRC 2006b, p. 11]. Assemblies manufactured for the production of tritium contained EU fuel elements along with the target elements. For example, the Mark VIII assembly contained uranium with 5%  $^{235}\text{U}$  [DuPont 1974a, p. 45].

Note that for EU, the degree of enrichment is not critical to dose reconstruction when the activity is known. Dose reconstructors should assume for uranium with any degree of enrichment, including TRM, that the activity represented is 100%  $^{234}\text{U}$ , the most favorable-to-claimant activity fraction. For cases where DU and EU might both be present and indistinguishable, both should be assessed. Assumed activity fractions for uranium enrichments might be needed to adjust isotopic uranium results to the total uranium activity for individual bioassay results. Uranium activity fractions for EU, DU, NU, and TRM are listed in Tables 5-8 to 5-11.

SRS technical documentation indicates that for earlier monitoring periods the designations enriched and depleted analysis for uranium referred to analysis performed by alpha counting or chemical measurement, respectively, and were not necessarily indicative of the degree of uranium enrichment. "EU" was the code used on employee bioassay cards for the gross alpha count method, and "DU" was used to designate the fluorophotometric method.

Results of uranium bioassay by DNC from 1982 to 1985 are for both "U" and "EU." Results for "EU" were reported in units of dpm/1.5 L, while results for "U" were sometimes reported in units of dpm/1.5 L and sometimes in units of  $\mu\text{g}/1.5\text{ L}$ . Handwritten Kardex bioassay records typically do not include units of the results in this era, so a Kardex result for "U" during this era could be in units of either dpm/1.5 L or  $\mu\text{g}/1.5\text{ L}$ . When the units for uranium are not provided in the worker's records from (approximately) 1982 to 1985, a result without units should be assumed to be in units of dpm because it provides a higher radioactivity concentration with a DU specific activity of 0.372 pCi/ $\mu\text{g}$  (0.826 dpm/ $\mu\text{g}$ ).

## Recycled Uranium

With the exception of incoming NU before 1961, all uranium at SRS was recycled and contained activation and FP impurities. Uranium extracted during fuel reprocessing, converted to  $UO_3$ , then returned to the offsite gaseous diffusion plants had low levels of radioactive impurities. Starting in 1961, this uranium was returned to the FMPC [McCarty 2000, p. 83] along with recycled uranium from other DOE facilities [DOE 2000, p. 18]. Uranium was received from the FMPC in 1961 and later was assumed to be recycled [McCarty 2000, p. 6]. NU used in SRS reactors as fuel or processed in canyon contained impurities as well. While there is no source data for those activity concentrations, the activity concentrations of impurities of NU would fall within the range in Table 5-26 for EU and DU. Since the activity concentrations reported for DU are high, assume those values for NU as well. Dose reconstructors should add these impurities to intakes of uranium by SRS workers based on the ratios in Table 5-26 for 300-M Area workers during 1961 and later and for all other uranium locations during 1953 and after. Impurities otherwise assigned from bioassay, including missing dose intakes, should not be added. The remaining impurities should still be included. TRM from the Chalk River Reactor in Canada may be assumed to have similar impurities to SRS reactors and EU values should be applied. Note that the reactor design at SRS was based partially on the Chalk River Reactor [Strack 2002, pp. 71–73].

Table 5-26. Ratios of impurities to total uranium (pCi/pCi U).<sup>a,b</sup>

Type	Pu-238	Pu-239/240	Np-237	Pu-241	Tc-99	Ru-106
EU	2.19E-06	3.58E-08	4.15E-04	2.77E-06	1.36E-04	5.29E-06
DU	1.93E-05	8.26E-04	3.67E-06	7.95E-03	4.90E-02	5.81E-04

Type	Eu-155	Am-241	Cs-137	Sb-125	Sr-90
EU	3.63E-04	3.23E-08	5.43E-05	N/A	3.00E-05
DU	1.28E-05	7.33E-05	3.97E-04	1.62E-03	2.94E-04

a. Source: Crase and LaBone [2000, pp. 7–8].

b. N/A = not applicable.

HP area codes are found in certain external dosimetry reports and indicate dosimetry badge locations [ORAUT 2024a, Section 2.2.18.1, Health Physics Area Codes]. Tables 5-27 through 5-30 and Table 5-32 list the HP Area Codes for the uranium locations. For these and other locations, contamination from past operations should be assumed by the dose reconstruction through the date of decontamination or demolition.

Table 5-27. HP Area codes for locations where DU was used.<sup>a</sup>

Period	H-Canyon and A-Line (1955–present) <sup>b</sup>	H-Area unknown facility (1955–present) <sup>b</sup>	Old HB-Line Facility (221-H) <sup>b</sup> (1955–1984)	723-A, 773-A <sup>c</sup> (1955–present)	735-A and 735-11A <sup>c</sup> (1955–present)	776-A <sup>c</sup> (1955–present)	777-M <sup>d</sup> (1955–1988)	Not identifiable or unknown <sup>e</sup> (1955–present) <sup>b</sup>	Reactors (R, P, L, K, C) (1955–1988) <sup>f</sup>	235-F Vaults	772-F and 772-1F Labs (1955–2021) <sup>g</sup>	F-Canyon A-Line, FB-Line (221-F), and F-Area unknown facility <sup>b</sup> (1955–2006) <sup>h</sup>	F-Area Outside Facilities <sup>h</sup> (1955–2006) <sup>i</sup>	320-M, 321-M <sup>f</sup> Unknown facility (1955–1988) <sup>j</sup>	CMX [679-T] (09/01/1951–1983) and TNX [678-T] (12/10/1951–09/2004) <sup>k</sup> (1955–1988)	Central Shops (including CTW position codes) (1955–present) <sup>l</sup>
1959–1972	2A	2A	2A	6N, 5A	6F	5A (773-A)	5B	None	7A, 8A, 9A, 10A, 11A	1A	1A <sup>h</sup>	1A	1B	3A	5C	1Z, 12D, 12E, 12Z
1973–08/16/1990	1H, 2H	1H through 4H, 7H	1H, 2H	7L, 5A	5D	5A (773-A)	5B	7Y, 8D, 8E	1C, 3C, 4C, 6C, 1K, 1P, 1L, 1R	2F, 5F, 8F	1A <sup>h</sup>	1F, 2F	9F	3M	5C	3F, 5H, 5F, 5W, 6H, 7A, 7B, 7J, 7R, 7T, 8A, 8B, 8C, 8H, 8I, 8M, 8P, 8S, 8T, 8K, 8L, 7Y
08/17/1990–2003	H01, H02	H01 through H04, H07	NA	A27, A15	A16	A15 (773-A)	Not applicable	J25, J26, Y01	C01, C03, C04, C06, K01, L01, P01, R01	2F, F05, F08	A01	F01, F02	F09	M03	T01	A18, F01, F03, F05, H05, H06, J01, J02, J03, J05, J06, J08, J10, J12, J14, J17, J18, J20, J21, J24, J30, J31, Y01
2004–present	HCA	299, HBL, HCA, EPT, HTF	NA	SRT, SRE	SRT (apply 773-A intakes)	SRT (apply 773-A intakes)	SRT	Blank, any code not already listed	LLL, NMM, SDD <sup>m</sup>	235	CLB	235, CLB, FCA FBL	FCA	SDD <sup>m</sup>	No active codes	Use occupation given in the computer-assisted telephone interview and/or DOL records

a. Source: ORAUT [2014a].

b. Separated plutonium from DU target material using PUREX process [SRNS 2013b, pp 181–182, 185].

c. The A-Area (700 Area with building numbers in the 700s) also contains the SRNL (Building 773-A), Main Laboratory Buildings (735-A and 735-11A), and the Liquid Waste Handling Facility (Building 776-A). All are still operating [ORAUT 2024a, Tables 2-47, 2-48, and 2-49].

d. Contained the Physics Laboratory with three test reactors: the Process Development Pile, the Standard Pile, and the Subcritical Experiment test reactors that were used to test the fuel and targets manufactured in the 300 Area [WSRC 2006b, pp. 37, 48, 113]. The reactors in 777-M were effectively shut down in the 1980s after most of their functions had been assumed by computer modeling [Strack 2002, p. 413].

e. Unknown facility radionuclides should only be assigned if no information is available from any source about the worker's work location.

f. Irradiated DU targets to make plutonium [SRNS 2013b, p. 185].

g. The production control laboratories, 772-F and 772-1F, were near the 221-F Canyon Facility and provided analytical support for the 200-F and 200-H separations processes [SRNS 2013b, p. 151].

h. Code 1A was used for both 772 and 773 before 1991. If no other information about work location is available, the applicable radionuclides for both locations should be assigned.

i. Starting in 1968, separation of plutonium from irradiated DU target material [Severynse 1998, p. 7].

j. Received DU from Y-12 and FMPC. DU shipments from SRS began in 1955 [McCarty 2000, pp. 75, 84–85]; Manufactured uranium into fuel elements and into target elements [WSRC 2006b, p. 42].

- k. Process pilot plants that tested fuel and target elements. TNX construction began in May 1951 and was completed by the following fall. The first uranium had been introduced and pilot runs began December 10, 1951 [Strack 2002, p. 417]. TNX has continued to serve as a pilot plant for operations in the F and H Areas [Strack 2002, p. 418]. CMX construction began in April 1951, and the facility was turned over to operations in September of the same year [Strack 2002, p. 413]. After 1983, the testing of new fuel and target elements was moved from CMX to SRL [WSRC 2006b, p. 48].
- l. CTWs might have worked anywhere on site. The Ford Building at Central Shops was used to repair contaminated reactor heat exchangers since at least 1961 and the mid-1980s [DuPont 1961, p. 282].
- m. Code SDD is used both for the reactors and for 300-M Area. If no other information about work location is available, the applicable radionuclides for both locations should be assigned.

Table 5-28. HP Area codes for locations where EU was used.<sup>a</sup>

Period	723-A, 773-A <sup>b</sup> (05/1959–present)	735-A and 735-11A <sup>b</sup> (05/1959–present)	776-A <sup>b</sup> (05/1959–present) <sup>c</sup>	777-M (1968–1988) <sup>c</sup>	CMX [679-T] (09/01/1951–1983) and TNX [678-T] (12/10/1951–09/2004) <sup>d</sup> (1968–1988)	772-F and 772-1F Labs <sup>e</sup> (05/1959–present)	Reactors (R, P, L, K, C) (1968–1988) <sup>f</sup>	320-M, 321-M, M-Area unknown facility (1954–1988) <sup>g</sup>	H-Canyon–A-Line, HB-Line (221-H), Outside Facilities (05/1959–present) <sup>h</sup>	Central Shops (Including CTW position codes) (05/1959–present) <sup>i</sup>
1968–1972	6N, 5A	6F	5A (773-A)	5B	5C	1A <sup>j</sup>	7A, 8A, 9A, 10A, 11A	3A	2A	1Z, 12D, 12E, 12Z
1973– 08/16/1990	7L, 5A	5D	5A (773-A)	5B	5C	1A <sup>j</sup>	1C, 3C, 4C, 6C, 1K, 1P, 1L, 1R	3M	1H, 2H	3F, 5F, 5H, 6H, 5W, 7A, 7B, 7J, 7R, 7T, 8A, 8B, 8C, 8H, 8I, 8M, 8P, 8S, 8T, 8K, 8L, 7Y
08/17/1990 –2003	A27, A15	A16	A15 (773-A)	Not applicable	T01	A01	Not applicable	M03	H01, H02	A18, F01, F03, F05, H05, H06, J01, J02, J03, J05, J06, J08, J10, J12, J14, J17, J18, J20, J21, J24, J30, J31, Y01
2004– present	SRT, SRE	SRT (apply 773-A intakes)	SRT (apply 773-A intakes)	Not applicable	No active codes	CLB	Not applicable	SDD <sup>k</sup>	HCA, HBL	Use occupation given in the computer-assisted telephone interview and/or DOL records

- a. Source: ORAUT [2014a].
- b. The A-Area (700 Area with building numbers in the 700s) also contains the SRNL (Building 773-A), Main Laboratory Buildings (735-A and 735-11A), and the Liquid Waste Handling Facility (Building 776-A). All are still operating [ORAUT 2024a, Tables 2-47, 2-48, and 2-49].
- c. Contained the Physics Laboratory with three test reactors: the Process Development Pile, the Standard Pile, and the Subcritical Experiment test reactors that were used to test the fuel and targets manufactured in the 300 Area [WSRC 2006b, pp. 37, 48, 113]. The reactors in 777-M were effectively shut down in the 1980s after most of their functions had been assumed by computer modeling [Strack 2002, p. 413].
- d. Process pilot plants that tested fuel and target elements. TNX construction began in May 1951 and was completed by the following fall. The first uranium had been introduced and pilot runs began December 10, 1951 [Strack 2002, p. 417]. TNX has continued to serve as a pilot plant for operations in the F and H Areas [Strack 2002, p. 418]. CMX construction began in April 1951, and the facility was turned over to operations in September of the same year [Strack 2002, p. 413]. After 1983, the testing of new fuel and target elements was moved from CMX to SRL [WSRC 2006b, p. 48].
- e. The production control laboratories, 772-F and 772-1F, were near the 221-F Canyon Facility and provided analytical support for the 200-F and 200-H separations processes [SRNS 2013b, p. 151].
- f. Operated with EU fuel elements after 1968 [Phifer et al. 2008, p. 23].
- g. Received EU, DU, and NU from Y-12 and FMPC. NU shipments began in 1953, DU shipments from SRS began in 1955, and EU shipments began in 1954 [McCarty 2000, pp. 75, 84–85]; manufactured uranium into fuel elements and into target elements [WSRC 2006b, p. 42].
- h. Separated plutonium from EU after May 1959 using HM process [SRNS 2013b, p. 195].
- i. CTWs might have worked anywhere on site. The Ford Building at Central Shops was used to repair contaminated reactor heat exchangers since at least 1961 and the mid-1980s. [DuPont 1961, p. 282].
- j. Code 1A was used for both 772 and 773 before 1991. If no other information about work location is available, the applicable radionuclides for both locations should be assigned.
- k. Code SDD is used both for the reactors and for 300-M Area. If no other information about work location is available, the applicable radionuclides for both locations should be assigned.

Table 5-29. HP Area codes for locations where NU was used.<sup>a</sup>

Period	Reactors (R, P, L, K, C) (12/1953–1967) <sup>b</sup>	320-M, 321-M (12/1953–1967) <sup>b,c</sup>	H Canyon–A-Line, HB Line (221-H), Outside Areas <sup>c</sup> (11/1954–1967) <sup>d</sup>	777-M <sup>e</sup> (12/1953–1967)	CMX [679-T] (09/01/1951–1983) and TNX [678-T] (12/10/1951–09/2004) <sup>f</sup>	Central Shops (including CTW position codes) <sup>g</sup> (12/1953–1967)
1953–1967	7A, 8A, 9A, 10A, 11A	3A	2A	5B	5C	1Z, 12D, 12E, 12Z

- Source: ORAUT [2014a].
- Operated with NU fuel elements after startup [Phifer et al. 2008, p. 23].
- Received EU, DU, and NU from Y-12 and FMPC. NU shipments began in 1953, DU shipments from SRS began in 1955, and EU shipments began in 1954 [McCarty 2000, pp. 75, 84–85]; manufactured uranium into fuel elements and into target elements [WSRC 2006b, p. 42].
- Starting in November 1954, separated plutonium from irradiated NU target material utilizing the PUREX process. Starting in 1968, separation of plutonium from irradiated DU target material [Severynse 1998, p. 7].
- Contained the Physics Laboratory with three test reactors: the Process Development Pile, the Standard Pile, and the Subcritical Experiment test reactors that were used to test the fuel and targets manufactured in the 300 Area [WSRC 2006b, pp. 37, 48, 113]. The reactors in 777-M were effectively shut down in the 1980s after most of their functions had been assumed by computer modeling [Strack 2002, p. 413].
- Process pilot plants that tested fuel and target elements. TNX construction began in May 1951 and was completed by the following fall. The first uranium had been introduced and pilot runs began December 10, 1951 [Strack 2002, p. 417]. TNX has continued to serve as a pilot plant for operations in the F and H Areas [Strack 2002, p. 418]. CMX construction began in April 1951, and the facility was turned over to operations in September of the same year [Strack 2002, p. 413]. After 1983, the testing of new fuel and target elements was moved from CMX to SRL [WSRC 2006b, p. 48].
- CTWs might have worked anywhere on site. The Ford Building at Central Shops was used to repair contaminated reactor heat exchangers since at least 1961 and the mid-1980s [DuPont 1961, p. 282].

Table 5-30. HP Area codes for locations where TRM was used.<sup>a,b</sup>

Period	H Canyon–A-Line, HB Line (221-H), Outside Areas <sup>b</sup>	772-F, 772-1F <sup>c</sup>	723-A, 773-A <sup>d</sup>	735-A, 735-11A <sup>d</sup>	776-A <sup>d</sup>	Central Shops (including CTW position codes) <sup>e</sup>
2017–2020	HCA, HBL	CLB	SRT	SRT (apply 773-A intakes)	SRT (apply 773-A intakes)	Use occupation given in the computer-assisted telephone interview and/or DOL records

- Source: ORAUT [2014a].
- Processed TRM between 2017 and 2020 [NNSA 2021, p. 3].
- The production control laboratories, 772-F and 772-1F, were near the 221-F Canyon Facility and provided analytical support for the 200-F and 200-H separations processes [SRNS 2013b, p. 151].
- The A-Area (700 Area with building numbers in the 700s) also contains the SRNL (Building 773-A), Main Laboratory Buildings (735-A and 735-11A), and the Liquid Waste Handling Facility (Building 776-A). All are still operating [ORAUT 2024a, Tables 2-47, 2-48, and 2-49].
- CTWs might have worked anywhere on site.

### 5.6.5.1 Uranium-233

SRS performed theoretical research and some development work on <sup>233</sup>U production proposed to be produced from the Mark VIB-VIIIT lattice in 1962. In 1963 the <sup>232</sup>U contamination considered permissible at that time was of the order of 50 ppm. Then in 1963, theoretical studies were initiated to examine various methods of producing <sup>233</sup>U with <sup>232</sup>U contents ranging from 10 ppm down to less than 1 ppm [DuPont 1984a, p. 70]. Thorium oxide irradiation to produce 130 kg of <sup>233</sup>U began in 1964 with about 6.5 ppm <sup>233</sup>U [DuPont 1984a, p. 94]. Different amounts of <sup>233</sup>U production within various levels of <sup>232</sup>U contamination occurred from 1964 through 1969 [DuPont 1965b, p. 8; 1984a, pp. 76, 80 (1964), 94 (1965), 106 (1966); 1984b, pp. 113–117 (1965), 138–139 (1966), 160 (1967), 181 (1968), 199 (1969)].

The HP area codes for the locations where <sup>233</sup>U was used are listed in Table 5-31. For unmonitored workers in those areas and periods, use the <sup>232</sup>U and <sup>233</sup>U intakes in ORAUT-TKBS-0003-7 [ORAUT 2024b, Tables 7-31 to 7-36]. In addition add the intakes from the radionuclides, from Table 5-32. For

Table 5-31, an evaluation for  $^{233}\text{U}$  should be performed. For the dose determination, include the additional nuclides based on Table 5-32 (this would include the  $^{232}\text{U}$  portion). Note that the  $^{233}\text{U}$  exposure potential was monitored using the “EU” bioassay method. See Section 5.3.2.4.

Table 5-31. HP Area codes for locations where  $^{232/233}\text{U}$  was used.<sup>a</sup>

Period	H Canyon (1964–1969) <sup>b</sup>	772-F, 772-1F Laboratories (1964–1969) <sup>c,d</sup>	773-A, 723-A, 776-A <sup>e</sup> (1964–1969) <sup>e,f</sup>	Central Shops (including CTW position codes) (1964–1969) <sup>g</sup>
1959–1972	2A	1A	5A	1Z, 12D, 12E, 12Z

a. Source: ORAUT [2014a].

b. Separated U-233 between 1964 and 1969 [Till 2001, p. 102].

c. Uranium-232/233 should only be assigned for 772-F and the Old HB-Line for January 1, 1964, through December 31, 1969. Thorium oxide irradiation to produce 130 kg of U-233 began in 1964 [DuPont 1984a, p. 94]. The Thorex IIB campaign in the plant was completed with the recovery of 181 kg of U-233 [DuPont 1984b, p. 193].

d. The production control laboratories, 772-F and 772-1F, were near the 221-F Canyon Facility and provided analytical support for the 200-F and 200-H separations processes [SRNS 2013b, p. 151]. Thorium oxide irradiation to produce 131 kg of U-233 in 1964 [DuPont 1984a, p. 94] The Thorex IIB campaign in the plant was completed in 1969 with the recovery of 181 kg of U-233 [DuPont 1984b, p. 193]. 772-F and 772-1F ended operations in 2021 [ORAUT 2024a, Table 2-21].

e. The A-Area (700 Area with building numbers in the 700s) also contains the SRNL (Building 773-A), Main Laboratory Buildings (735-A and 735-11A), and the Liquid Waste Handling Facility (Building 776-A). All are still operating [ORAUT 2024a, Tables 2-47, 2-48, and 2-49].

f. Uranium-232/233 should only be assigned for 773-A for 1964 through 1969. Laboratory work on the separation of U-233 from thorium was before operations which began in 1964 [DuPont 1984b, p. 10].

g. CTWs might have worked anywhere on site. The Ford Building at Central Shops was used to repair contaminated reactor heat exchangers since at least 1961 and the mid-1980s. [DuPont 1961, p. 282].

Table 5-32. Additional intakes to add with  $^{233}\text{U}$  intake (pCi/pCi).<sup>a</sup>

Nuclide	Intake to add
U-232	2.0E-2
Pu-239	1.3E-2
Th-232	1.0E-6
Np-237	2.1E-6
Zr/Nb-95	3.6E-5
Ru-106	4.8E-6
Pa-233	1.1E-2
Th-228	8.7E-5

a. Source: ORAUT [2020a, p. 44].

### 5.6.5.2 Uranium-236

SRS maintained MDAs for  $^{236}\text{U}$  bioassay but might not have reported bioassay results for  $^{236}\text{U}$  through 1989. The ICRP Publication 68 dose coefficients (also referred to as dose conversion factors) for  $^{234}\text{U}$  are equal to or greater than those for  $^{235}\text{U}$ ,  $^{236}\text{U}$ , and  $^{238}\text{U}$  [ICRP 1994]. When gross uranium activities are reported, because of the isotopic compositions of the source terms, the  $^{234}\text{U}$  dose conversion factor for gross uranium results yields doses that are favorable to claimants. Therefore, it is not necessary to consider any additional dose contribution from  $^{236}\text{U}$ . However, if isotopic results are reported, include the appropriate  $^{236}\text{U}$  activity fractions given in Tables 5-8 to 5-11 in the dose calculations.

### 5.6.5.3 Unmonitored Worker Uranium Dose

Internal uranium co-exposure intake rates as outlined in ORAUT-TKBS-0003-7 may be used for dose assignment when monitoring data are lacking (incomplete or unavailable) [ORAUT 2024b] and may be extended past 1990. Consult ORAUT-TKBS-0003-2 for rationale [ORAUT 2024a]. However, do not apply uranium co-exposure intake rates for SEC class members from October 1, 1972, through December 31, 1990.

### 5.6.6 Polonium-210

Reconstruct intakes of  $^{210}\text{Po}$  only for workers with reported  $^{210}\text{Po}$  results. Both Types F and M should be considered [ICRP 1994, p. 80].

### 5.6.7 Fission Products

As discussed in Section 5.3.2.6, gross beta or gross gamma urinalyses were used for monitoring FPs. ORAUT-RPRT-0047, as implemented in the OTIB-0054 tool, is used for interpretation of the results and dose assessment from all associated radionuclides in the mixture [ORAUT 2013b]. In cases where both gross beta and gross gamma results are available for a given sample, the gross beta results should be used because they are the more sensitive of the two methods. Table 3-3 in ORAUT-RPRT-0047 should be used to assign the  $^{90}\text{Sr}$  intake associated with a given gross beta urinalysis result [ORAUT 2013b]. Select "Major chemical processing" in the OTIB-0054 tool to apply these values. Intakes for the other radionuclides in the mix should then be assigned using Table 2-5 for the  $^{90}\text{Sr}$  indicator (i.e., gross beta).

FP urinalysis was generally not used in 1953 and 1954 when the reactors were first operating. Because tritium was the dominant source of intake at the reactors, if there was evidence that a worker had a tritium intake in 1954, then dose reconstructors should assume an intake using FP intake rates for 1955 in ORAUT-TKBS-0003-7 [ORAUT 2024b].

WB count data can be used to assess FP intakes. In this case,  $^{137}\text{Cs}$  is used as the indicator radionuclide for applying ORAUT-RPRT-0047 [ORAUT 2013b]. WB counts should be used for FP missed dose assessments from 1969 to 1989 rather than FP urinalysis results. If WB counts are not available,  $^{137}\text{Cs}$  co-exposure intake rates should be used.

The OTIB-0054 tool decay times of 10 days, 180 days, and 1 year are applicable to SRS intakes; as noted in RPRT-0047, decay times of 40 days are not appropriate. Only the 1-year decay time is applicable after the reactors ceased operation [ORAUT 2013b].

#### 5.6.7.1 Unmonitored Worker Fission Product Dose

Internal FP co-exposure intake rates as outlined in ORAUT-TKBS-0003-7 with  $^{90}\text{Sr}$  or  $^{137}\text{Cs}$  as the indicator radionuclide may be used for dose assignment when monitoring data are lacking (incomplete or unavailable) and may be extended past 1990 [ORAUT 2024b]. FP co-exposure intakes should be assigned to unmonitored workers for work in D-Area from 1954 to 1999. Consult ORAUT-TKBS-0003-2 for rationale [ORAUT 2024a].

### 5.6.8 Neptunium-237

Dose reconstructors should use neptunium bioassay results to reconstruct neptunium doses through the present. Dose reconstructors should use neptunium co-exposure data to reconstruct doses for unmonitored workers in buildings identified in ORAUT-RPRT-0065, *An Evaluation of Neptunium Operations at Savannah River Site*, through 1989 [ORAUT 2016]. Those buildings and periods are given in Table 5-33. Assume Type M [WSRC 1990].



Table 5-33. HP Area codes for <sup>237</sup>Np.<sup>a</sup>

Period	H-Area unknown facility <sup>b</sup>	320-M, 321 M-Area, M Area unknown facility <sup>c</sup> (1961–1998)	735-A and 735-11A (1957–present)	777-M Standard Pile, Nuclear Test Gauge (1955, 1965–1968)	Not identifiable or unknown <sup>b</sup>	773-A, 776-A (1961–1988) and 723-A (1957–present)	235-F Vaults (1962–1983)	Old HB-Line (1963–1983) New HB-Line (1985–2011) H-Canyon & A-Line <sup>c</sup> 221-H Area Outside Facilities	Central Shops (includes CTW-specific HPA Area codes)
1959–1972	2A	3A	8F	5B	None	5A, 6N Not applicable	1A, 1F, 12E, 12Z	2A	12D, 12E, 12Z
1973–08/16/1990	1H through 4H, 7H	3M	5D	Not applicable	7Y, 8D, 8E, 000, missing (cannot find on HPA code list)	5A, 7L Not applicable	1F, 2F, 8F	1H, 2H	1A, 5F, 5H, 6H, 8F, 7A, 7B, 7G, 7I, 7J, 7K, 7L, 7M, 7N, 7R7R, 7Q, 7W, 8A through 8C, 8F, 8H through 8M, 8P, 8S, 8T
08/17/1990–2003	H01 through H04, H07	M03	Not applicable	Not applicable	J25, J26, Y01, missing (cannot find on HPA code list)	A15, A27	Not applicable	H01, H02	A18, F05, H05, H06, J01, J02, J03, J05, J06, J08, J10, J12, J14, J17, J18, J20, J21, J24, J30, J31, Y01
2004–present	299, HBL, HCA, EPT, HTF	No active codes	Not applicable	Not applicable	Blank, any code not already listed	SRE, SRT	Not applicable	HBL (HB-Line) Not applicable	Use occupation given in the computer-assisted telephone interview and/or DOL records

a. Source: ORAUT [2014a].

b. Note that HP Area Codes alone are not sufficient to isolate a worker to certain individual locations such as A-Line, and without other evidence radionuclides given for H Canyon should be assumed.

c. Unknown facility radionuclides should only be assigned if no information is available from any source about the worker's work location.

### 5.6.9 Thorium

Starting in 1990, SRS used in vivo counting as the principal bioassay method for thorium. SRS used results reported for the 239-keV photon peak (45%) of  $^{212}\text{Pb}$  to quantify  $^{228}\text{Th}$  [WSRC 1990, 1993a]. The chest counter, placed in operation in 1989, was configured with an array of four low-energy germanium detectors with thin carbide windows capable of measuring 15- to 400-keV photons [WSRC 2001]. Thorium intakes are determined by assuming the  $^{212}\text{Pb}$  is in secular equilibrium with  $^{228}\text{Th}$  and  $^{232}\text{Th}$ . Unless there is claim-specific information, triple-separated thorium is assumed because it results in intakes that are favorable to the claimant. The reported  $^{212}\text{Pb}$  activity is assumed to be equal to the  $^{228}\text{Th}$  activity so is used to assess a  $^{228}\text{Th}$  intake. An assumed ratio is then applied to determine the  $^{232}\text{Th}$  intake. ORAUT-OTIB-0076, *Guiding Reconstruction of Intakes of Thorium Resulting from Nuclear Weapons Programs* [ORAUT 2014b], provides additional information and details of how to perform the evaluation.

Thorium intakes apply to workers in locations in Building 773-A and 776-A (1972 to the present), 235-F (1972 to 1980), Central Shops (1972 to the present), and workers where work location is not determined. Reconstruct thorium doses from October 1972 to 1980 using the co-exposure guidance and rates in ORAUT-TKBS-0003-7 [ORAUT 2024b]. Reconstruct doses for routine work from 1981 to 1989 using an inhalation rate of 4.87 dpm/d and an ingestion rate of 0.30 alpha dpm/d as detailed in ORAUT-RPRT-0070, *Evaluation of Method for Assessment of Thorium-232 Exposures at the Savannah River Site from 1972 to 1989* [ORAUT 2017c]. For DuPont CTWs during this time (dismantling, construction, and decontamination work from 1981 to 1989), continue to use the 50th-percentile co-exposure intake rates in ORAUT-TKBS-0003-7 for thorium. Although included in the source data, the co-exposure results for CTWs do not apply to subcontractor CTWs between October 1, 1972, and December 31, 1990. For 1990 forward, reconstruct thorium doses only for workers known to have worked in locations with thorium and who have chest counts using either the reported value for the  $^{212}\text{Pb}$  peak or the MDA value 0.15 nCi. Assume triple-separated thorium for all periods using guidance given in ORAUT-OTIB-0076 [ORAUT 2014b].

### 5.6.10 Potential for Internal Exposure (No External or In Vitro Monitoring)

Starting approximately in the mid-1990s, the presence of WB count (FASTSCAN) reports do not necessarily indicate potential exposure when there are no matching external or in vitro monitoring data. A combination of the worker's job, location, and other monitoring data should be used to determine the potential exposure. Workers could have received internal monitoring (Canberra FASTSCAN) for reasons other than identification of internal exposures (such as a training or qualification requirement). Therefore, with consideration of job duties and worker location, for a period when a worker received routine WB counts (Canberra FASTSCAN) but was not externally monitored and did not receive in vitro monitoring, the assignment of ambient dose for potential external exposure and environmental intakes for potential internal exposure is appropriate.

### 5.6.11 Europium-152 (Positive Whole Body Counts)

When a positive WB count records a  $^{152}\text{Eu}$  result above a given MDA or TBD MDA, consider the following before assessing it for dose:

- If the count is stamped with "K-40" only, it is safe to assume zero for  $^{152}\text{Eu}$ .
- Europium-152 emits a wide range of gammas.
- It is a good calibration source but results in more opportunity for it to be mistakenly identified.

As appropriate to those considerations, assume there was a potential for an intake of  $^{152}\text{Eu}$  if there is an accompanying urine sample result for  $^{152}\text{Eu}$ . If several peaks for  $^{152}\text{Eu}$  are identified in the spectrum, further review of the result is warranted. Contact the Principal Scientist for Internal Dosimetry for guidance.

## 5.7 UNCERTAINTY AND UNMONITORED DOSE

### 5.7.1 Unmonitored Worker

For assignment of intakes to unmonitored workers, dose reconstructors should also consider guidance in ORAUT-OTIB-0014, *Technical Information Bulletin: Assignment of Environmental Internal Doses for Employees Not Exposed to Airborne Radionuclides in the Workplace* [ORAUT 2004b]. Internal co-exposure intake rates from ORAUT-TKBS-0003-7 may be used for dose assignment when monitoring data are lacking (incomplete or unavailable) [ORAUT 2024b].

### 5.7.2 Calculating Minimum Detectable Activity from Uncertainty

If an MDA is not reported with a measurement but the uncertainty ( $\sigma$ ) is reported (as often seen on WB reports), the MDA for the measurement can be calculated using this formula:

$$MDA = 3.29\sigma \quad (5-7)$$

where

$MDA$  = minimum detectable activity (nCi)

Note that the uncertainty is often expressed as %Error in relation to the reported Activity. It can be in terms of 1 or 2 SDs (usually denoted as "1 SD" or "2 SD"), and the value must be converted to units of activity consistent with the reported activity units. An example of a  $^{141}\text{Ce}$  WB counting result reported with a percent error at 2 SD is shown in Figure 5-13. Equation 5-8 shows the calculation using the activity (12.9 nCi) and percent error at 2 SD (54.7).

Nuclide Results for STD/PSSUM			
Nuclide	Activity (nCi)	%Error (2 SD)	Comments
K-40	96.3	22.4	
Ce-141	12.9	54.7	
totals:	109.		

Figure 5-13. Portion of WB count report, 1998 [ORAUT 2023].

$$\begin{aligned}
 MDA &= 3.29 \left( \frac{\text{error}}{2 \times 100} \right) \times \text{Ce-141 activity} \\
 &= 3.29 \left( \frac{54.7}{2 \times 100} \right) \times 12.9 \text{ nCi} \\
 &= 11.6 \text{ nCi}
 \end{aligned} \quad (5-8)$$

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## GLOSSARY

### 3 wt% plutonium, 6 wt% plutonium, and 12 wt% plutonium

SRS designations for typical mixtures of plutonium that indicate the mass fraction of  $^{240}\text{Pu}$  in the mixture.

### absorption type

Categories for materials according to their rates of absorption from the respiratory tract to the blood, which replaced the earlier inhalation clearance classes. Defined by the International Commission on Radiological Protection, the absorption types are F: deposited materials that are readily absorbed into blood from the respiratory tract (fast solubilization), M: deposited materials that have intermediate rates of absorption into blood from the respiratory tract (moderate rate of solubilization), and S: deposited materials that are relatively insoluble in the respiratory tract (slow solubilization).

### barn

Metric unit of area equal to  $1 \times 10^{-28}$  square meter used in high-energy physics to express the cross-sections of any scattering process.

### beta dose

Designation (i.e., beta) on some records for external dose from beta and less-energetic X-ray and gamma radiation, often for shallow dose or dose to the lens of the eye.

### beta radiation

Charged particle emitted from some radioactive elements with a mass equal to 1/1,837 that of a proton. A negatively charged beta particle is identical to an electron. A positively charged beta particle is a positron.

### curie (Ci)

Traditional unit of radioactivity equal to 37 billion ( $3.7 \times 10^{10}$ ) becquerels, which is approximately equal to the activity of 1 gram of pure  $^{226}\text{Ra}$ .

### depleted uranium (DU)

Uranium with a percentage of  $^{235}\text{U}$  lower than the 0.7% found in natural uranium. As examples, spent (used) fuel elements, byproduct tails, residues from uranium isotope separation, and some weapons materials contain depleted uranium. Depleted uranium can be blended with highly enriched uranium to make reactor fuel or used as a raw material to produce plutonium. Depleted uranium was used for  $^{239}\text{Pu}$  production at SRS, and the listed isotopic activity fractions were:

<u>Isotope</u>	<u>Activity fraction</u>
$^{234}\text{U}$	0.0840
$^{235}\text{U}$	0.0145
$^{238}\text{U}$	0.9015

The specific activity of depleted uranium with these fractions is 523.5 picocuries per microgram.

### dosimeter

Device that measures the quantity of received radiation, usually a holder with radiation-absorbing filters and radiation-sensitive inserts packaged to provide a record of absorbed dose received by an individual.

**dosimetry**

Measurement and calculation of internal and external radiation doses.

**dosimetry system**

System for assessment of received radiation dose. This includes the fabrication, assignment, and processing of external dosimeters, and/or the collection and analysis of bioassay samples, and the interpretation and documentation of the results.

**exposure**

(1) In general, the act of being exposed to ionizing radiation. (2) Measure of the ionization produced by X- and gamma-ray photons in air in units of roentgens.

**fast neutron**

Neutron with energy equal to or greater than 10 kiloelectron-volts. This type of neutron causes fission in some isotopes (e.g.,  $^{238}\text{U}$ ,  $^{239}\text{Pu}$ ).

**gamma radiation**

Electromagnetic radiation (photons) that originates in atomic nuclei and accompanies many nuclear reactions (e.g., fission, radioactive decay, and neutron capture). Gamma photons are identical to X-ray photons; the difference is that X-rays do not originate in the nucleus.

**highly enriched uranium**

Uranium enriched to at least 20%  $^{235}\text{U}$  for use as fissile material in nuclear weapons components and some reactor fuels. Also called high-enriched uranium. SRS lists the isotopic activity fractions for fuel for nuclear-powered submarines as:

<u>Isotope</u>	<u>Activity fraction</u>
$^{234}\text{U}$	0.9806
$^{235}\text{U}$	0.0194
$^{238}\text{U}$	0.0000

The specific activity of highly enriched uranium with these fractions is 6,207 picocuries per microgram.

**neutron**

Basic nucleic particle that is electrically neutral with mass slightly greater than that of a proton. There are neutrons in the nuclei of every atom heavier than normal hydrogen.

**nuclear track emulsion, Type A (NTA)**

Film made by the Eastman Kodak Company that is sensitive to fast neutrons. The developed image has tracks caused by neutrons that become visible under oil immersion with about 1,000-power magnification. The number of tracks in a given area is a measure of the dose from that radiation.

**photon**

Quantum of electromagnetic energy generally regarded as a discrete particle having zero rest mass, no electric charge, and an indefinitely long lifetime. The entire range of electromagnetic radiation that extends in frequency from  $10^{23}$  cycles per second (hertz) to 0 hertz.

**radiation**

Subatomic particles and electromagnetic rays (photons) with kinetic energy that interact with matter through various mechanisms that involve energy transfer.

**radioactivity**

Property possessed by some elements (e.g., uranium) or isotopes (e.g.,  $^{14}\text{C}$ ) of spontaneously emitting energetic particles (electrons or alpha particles) by the disintegration of their atomic nuclei.

**recycled uranium**

Uranium first irradiated in a reactor, then recovered through chemical separation and purification. Recycled uranium contains minor amounts of transuranic material (e.g., plutonium and neptunium) and fission products (e.g., technetium) or uranium products (e.g.,  $^{236}\text{U}$ ) after purification.

**rem**

Traditional unit of radiation dose equivalent that indicates the biological damage caused by radiation equivalent to that caused by 1 rad of high-penetration X-rays multiplied by a quality factor. The sievert is the International System unit; 1 rem equals 0.01 sievert. The word derives from roentgen equivalent in man; rem is also the plural.

**Special Product (S.P.)**

Neptunium-237.

**U.S. Atomic Energy Commission (AEC)**

Federal agency created in 1946 to assume the responsibilities of the Manhattan Engineer District (nuclear weapons) and to manage the development, use, and control of nuclear energy for military and civilian applications. The U.S. Energy Research and Development Administration and the U.S. Nuclear Regulatory Commission assumed separate duties from the AEC in 1974. The U.S. Department of Energy succeeded the U.S. Energy Research and Development Administration in 1979.

**whole body (WB) dose**

Dose to the entire body excluding the contents of the gastrointestinal tract, urinary bladder, and gall bladder and commonly defined as the absorbed dose at a tissue depth of 10 millimeters (1,000 milligrams per square centimeter). Also called penetrating dose.

**X-ray radiation**

Electromagnetic radiation (photons) produced by bombardment of atoms by accelerated particles. X-rays are produced by various mechanisms including bremsstrahlung and electron shell transitions within atoms (characteristic X-rays). Once formed, there is no difference between X-rays and gamma rays, but gamma photons originate inside the nucleus of an atom.