SEC Petition Evaluation Report
Petition SEC-00103

Report Rev #: 0 Report Submittal Date: 11/14/2008

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Site Expert(s):	None

Petition Administrative Summary			
Petition Under Evaluation			
Petition #	Petition Type	Petition Qualification Date	DOE/AWE Facility Name
SEC-00103	83.13	March 4, 2008	Savannah River Site

Petitioner Class Definition
Construction workers and all other workers in all locations at the Savannah River Site, Aiken, SC, from 01/01/1950 to present.

Class Evaluated by NIOSH
All construction workers who worked in any area at the Savannah River Site during the period January 1, 1950 through December 31, 2007.

NIOSH-Proposed Class(es) to be Added to the SEC
None. The thorium operations prior to 1960 remain reserved as the evaluation continues

Related Petition Summary Information			
SEC Petition Tracking #(s)	Petition Type	DOE/AWE Facility Name	Petition Status
SEC00114	83.13	Savannah River Site	Merged

Related Evaluation Report Information	
Report Title	DOE/AWE Facility Name
None	

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Evaluation Report Summary: SEC-00103, SRS

This evaluation report by the National Institute for Occupational Safety and Health (NIOSH) addresses a class of employees proposed for addition to the Special Exposure Cohort (SEC) per the *Energy Employees Occupational Illness Compensation Program Act of 2000*, as amended, 42 U.S.C. § 7384 *et seq.* (EEOICPA) and 42 C.F.R. pt. 83, *Procedures for Designating Classes of Employees as Members of the Special Exposure Cohort under the Energy Employees Occupational Illness Compensation Program Act of 2000*.

Petitioner-Requested Class Definition

Petition SEC-00103, qualified on March 4, 2008, requested that NIOSH consider the following class: *Construction workers and all other workers in all locations at the Savannah River Site, Aiken, SC, from 01/01/1950 to present*

Class Evaluated by NIOSH

Based on its preliminary research, NIOSH modified the petitioner-requested class. NIOSH evaluated the following class: All construction workers who worked in any area at the Savannah River Site during the period January 1, 1950 through December 31, 2007.

NIOSH-Proposed Class(es) to be Added to the SEC

Based on its full research of the class under evaluation, NIOSH has obtained worker monitoring data, area and air monitoring data, source term information, and process information that allow dose reconstruction to be performed with sufficient accuracy. Based on its analysis of these available resources, NIOSH found no part of the class under evaluation for which it cannot bound radiation doses with sufficient accuracy. However, NIOSH has reserved the feasibility determination for thorium exposures from January 1, 1950 through December 31, 1959; NIOSH is continuing to evaluate the thorium bounding approach for this time period.

Feasibility of Dose Reconstruction

Per EEOICPA and 42 C.F.R. § 83.13(c)(1), NIOSH has established (with the exception of the portion of the proposed class where the decision is reserved) that it has access to sufficient information to: (1) estimate the maximum radiation dose, for every type of cancer for which radiation doses are reconstructed, that could have been incurred in plausible circumstances by any member of the class; or (2) estimate radiation doses of members of the class more precisely than an estimate of maximum dose. Information available from the site profile and additional resources is sufficient to document or estimate the maximum internal and external potential exposure to members of the proposed class under plausible circumstances during the specified period.

Health Endangerment Determination

Per EEOICPA and 42 C.F.R. § 83.13(c)(3), a health endangerment determination is not required because NIOSH has determined that it has sufficient information to estimate dose for the members of the proposed class.

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SEC Petition Evaluation Report for SEC-00103

1.0 Purpose and Scope

This report evaluates the feasibility of reconstructing doses for all construction workers who worked in any area at the SRS during the period January 1, 1950 through December 31, 2007. It provides information and analyses germane to considering a petition for adding a class of employees to the congressionally-created SEC.

This report does not make any determinations concerning the feasibility of dose reconstruction that necessarily apply to any individual energy employee who might require a dose reconstruction from NIOSH. This report also does not contain the final determination as to whether the proposed class will be added to the SEC (see Section 2.0).

This evaluation was conducted in accordance with the requirements of EEOICPA, 42 C.F.R. pt. 83, and the guidance contained in the Office of Compensation Analysis and Support's (OCAS) *Internal Procedures for the Evaluation of Special Exposure Cohort Petitions*, OCAS-PR-004.

2.0 Introduction

Both EEOICPA and 42 C.F.R. pt. 83 require NIOSH to evaluate qualified petitions requesting that the Department of Health and Human Services (HHS) add a class of employees to the SEC. The evaluation is intended to provide a fair, science-based determination of whether it is feasible to estimate with sufficient accuracy the radiation doses of the class of employees through NIOSH dose reconstructions.[1]

42 C.F.R. § 83.13(c)(1) states: *Radiation doses can be estimated with sufficient accuracy if NIOSH has established that it has access to sufficient information to estimate the maximum radiation dose, for every type of cancer for which radiation doses are reconstructed, that could have been incurred in plausible circumstances by any member of the class, or if NIOSH has established that it has access to sufficient information to estimate the radiation doses of members of the class more precisely than an estimate of the maximum radiation dose.*

Under 42 C.F.R. § 83.13(c)(3), if it is not feasible to estimate with sufficient accuracy radiation doses for members of the class, then NIOSH must determine that there is a reasonable likelihood that such radiation doses may have endangered the health of members of the class The regulation requires NIOSH to assume that any duration of unprotected exposure may have endangered the health of members of a class when it has been established that the class may have been exposed to radiation during a discrete incident likely to have involved levels of exposure similarly high to those occurring during nuclear criticality incidents. If the occurrence of such an exceptionally high-level exposure has not been established, then NIOSH is required to specify that health was endangered for those workers

[1] NIOSH dose reconstructions under EEOICPA are performed using the methods promulgated under 42 C.F.R. pt. 82 and the detailed implementation guidelines available at http://www.cdc.gov/niosh/ocas.

who were employed for at least 250 aggregated work days within the parameters established for the class or in combination with work days within the parameters established for other SEC classes (excluding aggregate work day requirements).

NIOSH is required to document its evaluation in a report, and to do so, relies upon both its own dose reconstruction expertise as well as technical support from its contractor, Oak Ridge Associated Universities (ORAU). Once completed, NIOSH provides the report to both the petitioner(s) and to the Advisory Board on Radiation and Worker Health (Board). The Board will consider the NIOSH evaluation report, together with the petition, petitioner(s) comments, and other information the Board considers appropriate, in order to make recommendations to the Secretary of HHS on whether or not to add one or more classes of employees to the SEC. Once NIOSH has received and considered the advice of the Board, the Director of NIOSH will propose a decision on behalf of HHS. The Secretary of HHS will make the final decision, taking into account the NIOSH evaluation, the advice of the Board, and the proposed decision issued by NIOSH. As part of this decision process, petitioners may seek a review of certain types of final decisions issued by the Secretary of HHS.[2]

3.0 SEC-00103 Savannah River Site Class Definitions

The following subsections address the evolution of the class definition for SEC-00103, Savannah River Site (SRS). When a petition is submitted by a claimant, the requested class definition is evaluated as submitted. If the available site information and data justify a change in the petitioner's class definition, NIOSH will specify a modified class to be fully evaluated. After a complete analysis, NIOSH will determine whether to propose a class for addition to the SEC and will specify that proposed class definition.

3.1 Petitioner-Requested Class Definition and Basis

Petition SEC-00103, qualified on March 4, 2008, requested that NIOSH consider the following class for addition to the SEC: *Construction workers and all other workers in all locations at the Savannah River Site, Aiken, SC, from 01/01/1950 to present.*

The petitioner provided information and affidavit statements in support of the petitioner's belief that accurate dose reconstruction over time is impossible for the Savannah River Site workers in question based on "radiation exposures and radiation doses potentially incurred by members of the proposed class were not monitored either through personal monitoring or through area monitoring." The SEC00103 petitioners asserted the following in the SEC petition:

> "Since the inception of the EEOICP [sic], the building trades have asked NIOSH to come up with a unique approach to construction worker dose reconstructions that will take into account the unique employment patterns and unreliable dose monitoring. To date, NIOSH has failed to do so.
>
> In 2005, a study was performed by the Center to Protect Workers' Rights which has been provided to NIOSH compared 2,335 construction workers, who had been employed at the SRS site and who

[2] See 42 C.F.R. pt. 83 for a full description of the procedures summarized here. Additional internal procedures are available at http://www.cdc.gov/niosh/ocas.

had participated in the Former Worker Medical Screening Program for the SRS site, to the radiation dose records data set for the SRS site (known as HPAREH). A significant number of SRS construction workers have either no deep dose or all recorded "zero" doses in HPAREH. Based on HPAREH database of radiation monitoring records from SRS, it appears that underlying dose data are deficient for 50-90% of the construction workers employed at SRS. NIOSH has not explained how it can complete dose reconstructions in light of this deficiency. On May 10, 2003, NIOSH issued a site profile document for the SRS site which aimed to provide methods for dose reconstruction where individual worker monitoring records were deficient. Construction workers who had extensive employment experience from all phases of the SRS site operation met with NIOSH in Augusta on November 11, and identified deficiencies in the site profile document as it related to construction workers in a number of areas. Their concerns were also presented to the NIOSH Board on Radiation and Worker Health [sic] on December 9, 2003, to make sure there is a record of them at NIOSH. The opening comments from the Building Trades November 11th meeting are attached, as are comments made to the Advisory Board on December 9, 2003. The SRS site profile was revised a number of times in 2004 and 2005, but none of these modifications included the concerns raised by the building trades.

There is no recent evidence to suggest that there is any reason to have confidence in the dose reconstructions performed by NIOSH. In a Congressional hearing on October 23, Mr. Shelby Hallmark of the Department of Labor testified that in 2007 DOL had returned 2,811 dose reconstruction cases for re-work, due to deficiencies identified in the work that NIOSH had performed. After re-working these cases, 385 cases which had been denied were approved. In other words, 14% had been wrong the first time around. Further, Mr. Hallmark stated that DOL would soon send another 4,400 cases back to NISOH [sic], and in addition to that 5,000 more. This means that DOL will have sent back half of the dose reconstruction cases completed.

We conclude that in the six years that have elapsed since this program was implemented, NIOSH does not have a valid method to perform dose reconstructions for construction workers, and has not acted to rectify the deficiencies identified in the underlying knowledge base for the SRS site.

Therefore, we believe that dose reconstructions on SRS construction workers cannot be performed with the reliability intended by the Act, and therefore, the construction workers employed at the SRS site should be included in the SEC."

NIOSH has concluded that there is sufficient information and documentation, and a defined dose reconstruction method (included in the SRS site profile document, ORAUT-TKBS-0003), for all other SRS non-construction workers. The dose reconstruction approach in ORAUT-TKBS-0003 for all SRS non-construction workers, coupled with the available personnel monitoring data, supports NIOSH's ability to bound the dose for all SRS non-construction workers. Based on this information, NIOSH finds that there is insufficient support for the petition basis for SRS non-construction workers.

In addition, based on its SRS research and data capture efforts, NIOSH determined that it has access to worker, co-worker, area and air radiological monitoring and source term data for SRS construction trade workers during the time period under evaluation. However, NIOSH considered the information and statements provided by the petitioner sufficient to qualify the petition for further consideration by NIOSH, the Board, and HHS. The details of the petition basis are addressed in Section 7.4.

3.2 Class Evaluated by NIOSH

Based on its preliminary research, NIOSH modified the petitioner-proposed class in accordance with a ruling by the HHS administrative review panel (Branche, 2008). Therefore, NIOSH defined the following class for further evaluation: All construction workers who worked in any area at the Savannah River Site during the period January 1, 1950 through December 31, 2007.

3.3 NIOSH-Proposed Class(es) to be Added to the SEC

Based on its research, NIOSH has obtained monitoring data, source term information, and process information that allow dose reconstruction to be performed with sufficient accuracy. Based on its analysis of these available resources, NIOSH found no part of the class under evaluation for which it cannot estimate radiation doses with sufficient accuracy. However, NIOSH has reserved the feasibility determination for thorium exposures from January 1, 1950 through December 31, 1959; NIOSH is continuing to evaluate the thorium bounding approach for this time period.

4.0 Data Sources Reviewed by NIOSH to Evaluate the Class

ATTRIBUTION: Section 4.0 and its related subsections were completed by Mike Mahathy, Oak Ridge Associated Universities. These conclusions were peer-reviewed by the individuals listed on the cover page. The rationales for all conclusions in this document are explained in the associated text.

NIOSH identified and reviewed numerous data sources to determine information relevant to determining the feasibility of dose reconstruction for the class of employees under evaluation. This included determining the availability of information on personal monitoring, area monitoring, industrial processes, and radiation source materials. The following subsections summarize the data sources identified and reviewed by NIOSH.

4.1 Site Profile Technical Basis Documents (TBDs)

A Site Profile provides specific information concerning the documentation of historical practices at the specified site. Dose reconstructors can use the Site Profile to evaluate internal and external dosimetry data for monitored and unmonitored workers, and to supplement, or substitute for, individual monitoring data. The Site Profile for Savannah River site is a single document with sections that provide process history information, information on personal and area monitoring, radiation source descriptions, and references to primary documents relevant to the radiological operations at the site.

As part of NIOSH's evaluation detailed herein, it examined the following TBDs for insights into SRS operations or related topics/operations at other sites:

- *TBD: Savannah River Site,* ORAUT-TKBS-0003; Rev. 02; April 5, 2005; SRDB Ref ID: 20176

- *Savannah River Site TBD Revisions,* OCAS-PER-030; Rev. 0; December 12, 2007; SRDB Ref ID: 38872

4.2 ORAU Technical Information Bulletins (OTIBs) and Procedures

An ORAU Technical Information Bulletin (OTIB) is a general working document that provides guidance for preparing dose reconstructions at particular sites or categories of sites. An ORAU Procedure provides specific requirements and guidance regarding EEOICPA project-level activities, including preparation of dose reconstructions at particular sites or categories of sites. NIOSH reviewed the following OTIBs and procedures as part of its evaluation:

- *OTIB: Maximum Internal Dose Estimates for Savannah River Site Claims*, ORAUT-OTIB-0001, Rev 00; July 15, 2003; SRDB Ref ID: 19407

- *OTIB: Dose Reconstruction from Occupationally Related Diagnostic X-Ray Procedures*, ORAUT-OTIB-0006, Rev. 03 PC-1; December 21, 2005; SRDB Ref ID: 20220

- *OTIB: Assignment of Environmental Internal Doses for Employees Not Exposed to Airborne Radionuclides in the Workplace*, ORAUT-OTIB-0014, Rev 00; June 22, 2004; SRDB Ref ID: 19432

- *OTIB: Interpretation of Dosimetry Data for Assignment of Shallow Dose*, ORAUT-OTIB-0017, Rev. 01; October 11, 2005; SRDB Ref ID: 19434

- *OTIB: Internal Dose Overestimate for Facilities with Air Sampling Programs*, ORAUT-OTIB-0018,, Rev. 01; August 9, 2005; SRDB Ref ID: 19436

- *OTIB: Use of Coworker Dosimetry Data for External Dose Assignment*, ORAUT-OTIB-0020, Rev. 01; October 7, 2005; SRDB Ref ID: 19440

- *OTIB: External Coworker Dosimetry Data for the Savannah River Site*, ORAUT-OTIB-0032; November 7, 2006; SRDB Ref ID: 29964

- *OTIB: Parameters to Consider When Processing Claims for Construction Trade Workers*, ORAUT-OTIB-0052; Rev.00 PC-1, January 16, 2007; SRDB Ref ID: 29978

- *OTIB: Internal Dose Reconstruction,* ORAUT-OTIB-0060, Rev 00, February 6, 2007, SRDB Ref ID: 29984

- *OTIB: Occupational X-Ray Dose Reconstruction for DOE Sites*, ORAUT-PROC-0061, Rev. 01; July 21, 2006; SRDB Ref ID: 29987

- *OTIB: Calculation of Dose from Intakes of Special Tritium Compounds,* ORAUT-OTIB-0066, Rev. 00; April 26, 2007; SRDB Ref ID: 31421

- *OTIB: Use of Claimant Data Set for Coworker Modeling*, ORAUT-OTIB-0075, Rev. 00; no date yet; SRDB Ref ID: To be published in 2008

4.3 Facility Employees and Experts

NIOSH interviewed SRS employees in order to obtain input regarding health physics practices, internal and external dosimetry programs, dose recording practices, and radiological incidents.

- Personal Communication, 2008a, *Personal Communication with Current Worker, Summary of Discussion of Selected Cohen & Associates SRS Site Profile Comments;* Telephone Interview by ORAU Team; August 14, 2006; SRDB Ref ID: 27056

- Personal Communication, 2008b, *Personal Communication with Former Worker, Use of NTA Neutron Personnel Dosimetry in 1955-1968 at SRS*; Telephone Interview by ORAU Team; March 9, 2008; SRDB Ref ID: 48459

- Personal Communication, 2008c, *Personal Communication with Current Worker,* S*RS Incident Database, Special Hazards Investigations Database, HPAREH Database and Intake Registry;* Telephone Interview by ORAU Team; April 3, 2008; SRDB Ref ID: 45062

- Personal Communication, 2008d, *Personal Communication with Engineer Concerning Thorium,* Telephone Interview by ORAU Team; August 12, 2008; SRDB Ref ID: Not yet released by DOE

- Personal Communication, 2008e, *Personal Communication with Metallurgical Lab Technician,* Telephone Interview by ORAU Team; August 11, 2008; SRDB Ref ID: Not yet released by DOE

- Personal Communication, 2008f, *Personal Communication with Electrician,* Telephone Interview ORAU Team; August 11, 2008; SRDB Ref ID: Not yet released by DOE

- Personal Communication, 2008g, *Personal Communication with Former Health Physics Manager,* Telephone Interview ORAU Team; August 11, 2008; SRDB Ref ID: Not yet released by DOE

- SEC Worker Outreach Meeting for the Savannah River Site, National Institute for Occupational Safety and Health; May 22, 2008, 1 PM; SRDB Ref ID: Released by DOE; not yet in SRDB

- SEC Worker Outreach Meeting for the Savannah River Site, National Institute for Occupational Safety and Health; May 22, 2008, 6 PM; SRDB Ref ID: Not yet released by DOE

4.4 Previous Dose Reconstructions

NIOSH reviewed its NIOSH OCAS Claims Tracking System (NOCTS) to locate EEOICPA-related dose reconstructions that might provide information relevant to the petition evaluation. Table 4-1 summarizes the results of this review. (NOCTS data available as of October 1, 2008)

Table 4-1: No. of Savannah River Claims Submitted Under the Dose Reconstruction Rule

Description	Totals
Total number of claims submitted for dose reconstruction	3264
Total number of claims submitted for energy employees who meet the definition criteria for the class under evaluation (Construction / Building Trades from January 1, 1950 through December 31, 2007)	1798
Number of dose reconstructions completed for energy employees who meet the definition criteria for the class under evaluation (i.e., the number of such claims completed by NIOSH and submitted to the Department of Labor for final approval).	1358
Number of claims for which internal dosimetry records were obtained for the identified years in the evaluated class definition	1467
Number of claims for which external dosimetry records were obtained for the identified years in the evaluated class definition	1474

NIOSH reviewed each claim to determine whether internal and/or external personal monitoring records could be obtained for the employee. Dose reconstructions have been completed based on actual monitoring data (data reported for over 80% of construction worker claims) and with the use of ORAUT-OTIB-0052.

4.5 NIOSH Site Research Database

NIOSH also examined the Site Research Database (SRDB) to locate documents supporting the evaluation of the proposed class. As of March 2008, there were about 600 technical documents in the SRDB pertaining to the Savannah River Site. In order to address the petitioner's concerns and issues, NIOSH undertook a much larger on-site data review from June through August 2008. As a result, approximately 500 additional documents have been captured and reviewed (or are in the process of being reviewed, as is the case for the pre-1960 thorium assessment). In addition to technical reports, NIOSH has collected: (1) all of the Quarterly External Dosimeter reports since 1958; (2) all of the bioassay logbooks which contain the individual urinalysis sample results; and (3) the Site Special Hazards Investigation Reports (Incident Reports).

Documents evaluated for relevance to this petition include historical background on the evolution of the site, process descriptions, radiological monitoring data (surface and air concentrations, personnel external and internal exposures), information on the radiological controls program as well as monthly reports, incident documentation, and epidemiological studies.

4.6 Other Technical Sources

- OCAS-TIB-006, *Interpretation of External Dosimetry Records at the Savannah River Site (SRS)*, Rev. 2; National Institute for Occupational Safety and Health (NIOSH); Cincinnati, Ohio; October 4, 2007; SRDB Ref ID: 35409

- OCAS-TIB-007, *Neutron Exposures at the Savannah River Site*, Rev. 01; National Institute for Occupational Safety and Health (NIOSH); Cincinnati, Ohio; October 15 2007; SRDB Ref ID: 35675

- Health Protection Annual Radiation Exposure History Database (HPAREH)

- Health Protection Radiation Exposure Database (HPRED)

- SRS Site Incident Database

- Internal Dose Registry

4.7 Documentation and/or Affidavits Provided by Petitioners

In qualifying and evaluating the petition, NIOSH reviewed the following documents submitted by the petitioners:

- Petition Attachment 1 – *NIOSH/Union SPD Meeting, November 11, 2003*, November 19,2007; OSA Ref ID: 104377
  - Includes a review of the SRS SPD by former SRS union Energy Employees.
  - Specifically discusses the review and assessment of the SRS SPD and makes a conclusion about the available personnel monitoring data.

- Petition Attachment 2 – *Statement from AFL-CIO Science Advisor to NIOSH and the Board, December 9, 2003*, November 19, 2007; OSA Ref ID: 104377
  - CPWR statements made regarding SRS construction workers and its evaluation of the EEOICPA dose reconstruction program, conflict of interest, and Rule conflicts.

- Petition Attachment 3 – *CPWR-NIOSH Meeting on Variance in Construction Worker Radiation Exposure Monitoring, July 27, 2005*, November 19, 2007; OSA Ref ID: 104377
  - CPWR statements made regarding SRS construction workers and its evaluation of the EEOICPA dose reconstruction program and dose reconstruction methodology.

- Petition Attachment 4 – *What is the best estimate of the daily ventilation rate for construction workers? July 28, 2005*, November 19, 2007; OSA Ref ID: 104377
  - CPWR statements made regarding SRS construction workers and its evaluation of the EEOICPA dose reconstruction program and dose reconstruction methodology. The attachment includes a PowerPoint presentation and a newspaper article.

- Affidavit, *"Non-specific claimed radiological situations by former Energy Employee*," November 19, 2007; OSA Ref ID: 104377
- Affidavit , "*Non-specific claimed radiological situations by survivor*," November 19, 2007 OSA Ref ID: 104377
- Affidavit, "*Semi-specific claimed contamination incidents by former Energy Employee – no data and no way to confirm*," November 19, 2007; OSA Ref ID: 104377
- Affidavit, "*Semi-specific claimed contamination incidents by former Energy Employee*," November 19, 2007; OSA Ref ID: 104377
- Affidavit, *"Semi-specific claimed radiological situations by former Energy Employee*," November 19, 2007; OSA Ref ID: 104377
- Affidavit, "*Non-specific claimed radiological situations by former Energy Employee*," November 19, 2007; OSA Ref ID: 104377
- Affidavit, "*Non-specific claimed radiological situations and chemical exposures by former Energy Employee*," November 19, 2007 OSA Ref ID: 104377
- Affidavit, "*Non-specific claimed radiological situations and beryllium exposures by former Energy Employee*," November 19, 2007 OSA Ref ID: 104377
- Affidavit, "*Semi-specific claimed radiological situations (contaminated railroad cross-ties) by former Energy Employee*," November 19, 2007 OSA Ref ID: 104377
- Affidavit, "*Non-specific claimed radiological situations by former Energy Employee*," November 19, 2007; OSA Ref ID: 104377
- Affidavit, "*Semi-specific claimed radiological situations by former Energy Employee*," November 19, 2007; OSA Ref ID: 104377
- Affidavit, "*Non-specific claimed radiological situations associated with dosimeters by former Energy Employee*," November 19, 2007; OSA Ref ID: 104377
- Affidavit, "*Non-specific claimed radiological situations by former Energy Employee*," November 19, 2007; OSA Ref ID: 104377
- Affidavit, "*Non-specific claimed radiological by a former Energy Employee*," November 19, 2007; OSA Ref ID: 104377

5.0 Radiological Operations Relevant to the Class Evaluated by NIOSH

The following subsections summarize both radiological operations at the SRS from January 1, 1950 through December 31, 2007 and the information available to NIOSH to characterize particular processes and radioactive source materials. From available sources NIOSH has gathered process and source descriptions, information regarding the identity and quantities of each radionuclide of concern, and information describing both the processes through which radiation exposures to construction workers may have occurred and the physical environment in which they may have occurred. Construction workers were used in all of the processes and operations discussed in Section 5.1, although some work was new construction not involving the presence of radiological materials (DPSP-55-454-2). The information included within this evaluation report is intended only to be a summary of the available information.

5.1 Savannah River Site Plant and Process Descriptions

<u>ATTRIBUTION</u>: *Section 5.1 and its related subsections were completed by James K. Alexander, Oak Ridge Associated Universities. These conclusions were peer-reviewed by the individuals listed on the cover page. The rationales for all conclusions in this document are explained in the associated text.*

In 1950, the Atomic Energy Commission (AEC) selected a site in southeast South Carolina for construction of nuclear reactor facilities and ancillary operations that would manufacture plutonium, tritium, and other materials needed for the assembly of thermonuclear weapons. It was roughly circular in shape, slightly over 20 miles in diameter, and covered approximately 375 square miles. The location was chosen because of its proximity to the Savannah River, a large water source needed to remove the heat generated by the reactors and as a source of relatively clean water for heavy water extraction.

The five heavy-water-moderated production reactors subsequently built on the site (nominally designated as the C, L, P, K, and R Reactors), the F and H "canyon" chemical processing areas, and the D-Area Heavy Water Production Facilities were spaced two to three miles apart along a roughly seven-mile diameter circle placed near the geometric center of the site. This configuration was chosen to facilitate security and public safety considerations (Bebbington, 1990).

Over the site's entire operational history, more than 1,000 separate facilities have been established, concentrated on only 10 percent of the total land area (DOE, 2008). A chronology of SRS development and the various production facilities is provided in Attachment 1.

The material production activities associated with routine operation of the SRS reactors began in December 1953 with the start-up of the R Reactor, and continued until 1964 when demand for plutonium and tritium had significantly decreased; this reactor was then shut down. These decreasing demand trends slowly continued and the C Reactor was shut down in 1985. The K and P reactors were shut down in 1988. Operation of the L Reactor continued sporadically until 1991, when it was also shut down. K Reactor was restarted briefly in 1992 to facilitate the addition of a new cooling tower, but it was placed on "standby" by 1993. By 1996, all five SRS production reactors were considered to be in permanent "cold shutdown" status; none have been operated since. Between 1953 and 1988, the SRS reactors produced about 36 metric tons of plutonium (SRS Highlights, 2000; Bebbington, 1990; DOE, 2008; DOE/DP-0137).

Currently, the SRS supports various DOE environmental restoration missions associated with waste management, radioactive material vitrification, special nuclear material storage, research and development, and technology transfer. The only remaining weapons program mission at SRS is the operation of the Replacement Tritium Facility (RTF) that began in 1994 (SRS, 2008). Construction workers have worked at all of the facilities in construction and maintenance operations.

5.1.1 K, L, P, C and R Reactors and Associated Facilities

The five SRS production reactors were all heavy-water-moderated designs ranging in power levels from 2,400 MW-thermal to 3,000 MW-thermal. They produced plutonium and tritium for use in nuclear weapons and, from time to time, produced other isotopes for various purposes.

The reactors were built in sequence so that the first-completed could be turned over for operations and started up while construction on the remaining (geographically separated) reactors continued. The reactor buildings were designed to prevent the release of direct radiation and/or significant quantities of radioactive materials. The reactor enclosures were designed to withstand possible enemy attack, which was compatible with advanced containment safety systems.

The SRS reactors differed substantially from the "graphite pile" reactors that had been previously built and operated at Hanford in that heavy water was used as both a neutron moderator and as the primary reactor coolant. Water from the Savannah River was used as the secondary coolant for all five of the reactors. The SRS reactor vessels were all made of stainless steel, about 16 feet in diameter, with precisely located holes at the top and corresponding pins at the bottom that held the reactor core lattice of fuel rods, target assemblies, control rods, safety rods, and instrument assemblies securely in place.

R Reactor achieved operating status first, in December 1953, and was shut down permanently in 1964 when the demand for the weapons reactor products began to decrease.

P Reactor was started in February 1954 and was shutdown in August 1988 for maintenance. In February 1991, it was placed in cold standby and was to be used to provide spare parts for L Reactor and K Reactor. However, this potential use was eliminated by the subsequent permanent shutdown of both the L and K Reactors.

L Reactor achieved operating status in August 1954 and was placed in cold standby in 1968. It was restarted in October 1985, after upgrading, and was shut down for maintenance and safety upgrades in August 1988. It was placed in warm standby in December 1991, to be put into operation as a backup to K Reactor, if necessary; but was subsequently shut down permanently.

C-Reactor achieved operating status in March 1955 and was shut down in 1985 for maintenance. It was placed in cold standby in 1987 when cracks were observed in the reactor vessel. It was subsequently placed in permanent shutdown status.

K Reactor achieved operating status in October 1954 and was shut down in August 1988 for maintenance. Initial steps to restart K Reactor began in December 1991. Successful power ascension testing was completed in July 1992. Following ascension testing, the reactor was taken offline to allow for the tie-in of a new cooling tower. The tie-in was completed and an operating permit was issued in December 1992. In 1993, the cooling tower was successfully tested; however, the reactor was never restarted. K Reactor was then placed in cold standby, but the official status was changed to cold shutdown in 1996.

In all of the reactors, some of the deuterium in the heavy water was converted to radioactive tritium, other radioactive materials sometime escape from the fuel elements by way of defects in the cladding. Accordingly, the operating crews in the reactor buildings were shielded from the high-level radiation within some of the reactor operating areas, and protected against airborne radioactive materials. The routine work spaces in the reactor buildings routinely occupied by workers, such as the offices, control rooms, change rooms, and shops, had an independent ventilation system. The reactor process areas, including the fuel element assembly rooms and spent fuel storage basins, utilized once-through ventilation and were maintained at lower air pressures than the non-process building areas. The process area ventilation air was normally discharged through a series of air filters and up relatively tall stacks.

An important aspect of the SRS reactor process area ventilating systems was that they were always online. They needed no emergency action initiator if an abnormal level of radioactivity was detected in the air within the reactor rooms. The ventilation systems continued to operate regardless of the operating or shutdown status of the reactor.

In the operation of the reactors, particularly during the charging and discharging of fuels and targets, there was inevitably some dilution of the heavy water by ordinary water present as humidity in the ventilating air. Efficiency of operation of the reactors depended on maintenance of the chemical and isotopic purity of the moderator. Accordingly, auxiliary equipment included ion exchangers for chemical purification and continuous vacuum distillation columns to remove light water-contaminated moderator for return to the 400-D area. (Bebbington, 1990; SRS Highlights, 2000).

5.1.2 Heavy Water Rework Facility (400-D Area)

U.S. scientists developed the Girdler Sulfide (GS) chemical exchange production process in the 1930s and 1940s, which was first demonstrated on a large scale at a production facility built in Dana, Indiana in 1945. Using the GS process, the heavy water content of ordinary water can be increased from about 0.015% to 15-20% by counter-current exchange of water and hydrogen sulfide gas, first through cold towers and then through hot towers (Bebbington, 1990). Heavy water separations operations using the Girdler process began at SRS in October 1952. The primary process used 144 120-foot tall heavy cylindrical towers ranging from 6.5 feet to 12 feet in diameter. A second stage plant, using vacuum distillation in smaller towers, was built at the same time as the GS facilities. A facility to generate steam for the heavy water distillation process, and 75-megawatts of electricity for use on the entire site, was also built in the 400-D Area in the early 1950s. The nominal production capacity of each of the two stages of the plant was designated at 240 tons of heavy water per year. This rate was attained in April 1953. By mid-1954, production was over 400 tons per year, and reached 500 tons per year in 1956 (DOE, 2008).

5.1.3 Defense Waste Processing Facility (DWPF)

The Defense Waste Processing Facility processes and immobilizes the alkaline radioactive high-level waste (HLW) sludge generated at SRS into a durable borosilicate glass suitable for long-term storage in a geological repository. Following a ten-year construction period beginning in 1983, and a three-year non-radioactive testing program, full-scale operations began in March 1996.

Roughly 36 million gallons of liquid high-level nuclear wastes are now stored at SRS in 49 underground carbon-steel tanks. This waste contains about 421 million curies of radioactivity, of which the vast majority will be vitrified at the DWPF. It is projected that the DWPF will produce more than 5,000 vitrified waste canisters by 2019. As of January 2008, DWPF had produced more than 2,000 filled waste canisters derived from over 2,000,000 gallons of HLW sludge (SRS Defense, 2007).

5.1.4 Saltstone Facility

Saltstone consists of two facility segments: the Saltstone Production Facility (SPF) and the Saltstone Disposal Facility (SDF). Construction of SPF and the first two waste disposal vaults were completed between February 1986 and July 1988. The facility started radioactive operations on June 12, 1990. Since that time, it has been operated on an intermittent, as-needed basis to immobilize and dispose of low-activity liquid waste from the SRS Effluent Treatment Project (ETP) that processes waste from the F and H Canyons and their respective tank farms. These solutions contain low-level radioactivity and heavy metal ions. Immobilization is accomplished by mixing the solution with flyash, cement, and slag in the SPF, and pouring it into large concrete vaults to harden in the SDF. Saltstone also includes facilities to accommodate low-activity waste originating from the SRS Deliquifaction, Dissolution, and Adjustment (DDA) process, the Actinide Removal Process/Modular Caustic Side Extraction Unit (ARP/MCU), and eventually, the Salt Waste Processing Facility (SWPF) scheduled for start-up in 2013 (SRS Saltstone, 2007).

5.1.5 Tritium Facilities

The SRS Tritium Facilities provide tritium processing capabilities needed for post-cold-war nuclear weapons production and non-weapon uses. The Tritium Facilities consist of three main active process areas.

The first area, built in the early 1950s, includes Building 232-H which is a 55,000 square foot facility that historically performed all tritium extraction and purification operations. Ancillary to Building 232-H are Building 233-H (35,000 square feet), used for loading and unloading; and Building 234-H (46,000 square feet), used for shipping and receiving functions.

The second area, called the H Area New Manufacturing Facility (HANM), began operations in 1994. It was significantly upgraded in 2004 to consolidate tritium processing and handling activities, improve safety, reduce costs, and better control environmental releases.

The third and newest process area, the Tritium Extraction Facility (TEF), located in the H Area, became fully operational in early 2007.

The nation's tritium production capability supporting nuclear weapons manufacturing was temporarily suspended in 1988 with the shutdown of the last heavy water reactor at Savannah River. In December 1998, a decision was made that new tritium for nuclear weapons would be procured from government-owned nuclear power reactors operated by the Tennessee Valley Authority (TVA). Subsequently, Tritium Producing Burnable Absorber Rods (TPBARs) were loaded into the reactor at the Watts Bar Nuclear Plant, located in east Tennessee. After being irradiated, the TPBARs are shipped to the Savannah River Site for processing at the TEF. The extracted tritium is then piped to the Tritium Loading Facility for further purification prior to shipment to Department of Defense (DOD) installations where stockpiled weapons can be replenished on an as-needed basis (SRS Spent, 2008).

5.1.6 300 M-Area Fabrication Facilities

Historically, the 300 M-Area was the location of facilities that produced fuel tubes of enriched uranium-aluminum alloy with aluminum cladding. The fuel tubes were combined into assemblies and loaded into the production reactors to produce Pu-238, Pu-239, Cf-252, and tritium.

The current mission for the M Area Facilities is to facilitate the safe management of waste materials originating from reactor shut-down activities, to treat previously-generated mixed low-level waste in the Liquid Effluent Treatment Facility (LETF), and to prepare for the transition of facilities to DOE's Environmental Management (EM) programs for decontamination and decommissioning (D&D).

Starting in January 1953, SRS canned thorium metal slugs for Hanford fuel tests in Buildings 313-M and 320-M. In 1954, SRS began R&D and a small-scale production process to can thorium metal slugs for inclusion in SRS reactor fuel. By 1955, SRS was performing inspection and acceptance testing of thorium metal slugs that had been canned by off-site vendors. In 1964, SRS installed vibratory-compaction equipment in 313-M to facilitate the production of thorium oxide (thoria) slugs.

Building 320-M was where the alloy fabrication process was conducted for target tubes and control rods (lithium-aluminum alloy cores with aluminum cladding). Fabrication operations have been discontinued and all lithium target tubes have been removed from the facility.

Building 322-M was the Reactor Materials Quality Metallurgical and Physical Testing Laboratory where both fissile and non-fissile materials were examined. The building vault is now used to store waste materials containing fissile material.

Buildings 330-M and 331-M were warehouses where depleted uranium cores and slugs were stored, but now contain only small amounts of residual waste materials. Building 340-M was a liquid waste-handling facility that supported the Building 322-M operations, but now contains only small amounts of residual contamination.

The Liquid Effluent Treatment Facilities (LETF) consisted of three main facilities: the Dilute Effluent Treatment Facility [DETF] (Building 341-M); the Interim Treatment and Storage Facility [ITSF], (Building 341-1M); and the Vendor Treatment Facility [VTF] (Building 341-8M). The DETF was built in 1982 and received the production wastewater formerly discharged to the M-Area settling basins and outfalls. The DETF contained batch systems that processed wastewater by pH adjustment, precipitation, and filtration. The ITSF was built in 1985 to store and prepare production solids formerly disposed of in the low-level waste burial grounds. The VTF was built in 1996 to convert the low-level mixed waste sludge from the ITSF into leach-resistant solid glass gems. D&D of all of the buildings/facilities in the LETF area has been completed (Arcano, 1994).

5.1.7 Receiving Basin for Offsite Fuel (RBOF) and L-Basin

The Receiving Basin for Offsite Fuel (RBOF), located in the H Area near the center of the SRS, was a spent-fuel storage pool for research reactor fuels. It became fully operational in 1964 and continued program or decommissioning operations through 2006. The mission of the RBOF was to:

- receive, handle, and store irradiated nuclear fuel elements from off-site power and research reactors, from foreign country reactors, and from on-site reactors.
- repackage nuclear fuel elements into containers and bundles for extended storage and/or shipment to on-site or off-site reprocessing facilities.
- handle, separate, and transfer wastes generated from nuclear fuel element storage.

From 1964 to 1988, the RBOF supported the SRS mission through the safe interim storage of irradiated nuclear fuel elements from U.S., off-site, and foreign reactors in support of nonproliferation policies. Plans were then implemented to remove the spent fuel from the RBOF and relocate this material to the L Area Disassembly Basin (L-Basin), a much larger facility. Following the completion of modifications in 1996, L-Basin received its first shipment of foreign spent fuel in February 1997. By October 2003, all the spent fuel previously stored in the K-Basin and the RBOF had been removed either to chemical separations facilities or the L-Basin, leaving the L-Basin as the only remaining SRS spent fuel receipt and storage facility.

5.1.8 F Canyon

F Canyon, one of two former chemical separations areas at SRS, is a 128,000 square foot plutonium and uranium separations facility previously used to process plutonium and other materials for national defense purposes. The facility was built in the early years of SRS operations and began chemical separations in November 1954. The operations area is 835 feet long, 122 feet wide, and 66 feet high, resembling a gorge in a deep valley between steeply vertical cliffs, hence the term "canyon."

To limit worker exposure to the relatively-high radiation fields in the canyon, work was remotely performed using overhead cranes. Thick, very dense concrete walls also separated workers from the radiation present in the actual interior processing areas.

F Canyon began phasing out its traditional production operations in the late 1990s and all operations were concluded by March 2002. The F Canyon operations included a plutonium purification cycle to concentrate Pu-239 for transfer to the FB Line; processing spent fuel rods from the Taiwan Research Reactor (TRR) and Mark-31 targets; storing other plutonium, uranium, and Am/Cm solutions awaiting restart authorization; and operating nondiscretionary waste evaporation cycles to process canyon, analytical laboratory, reactor, and related waste streams. The F Canyon Outside Area Facility, including the FA Line that was used for uranium processing, was a 37,500 square foot complex providing support operations.

While awaiting a decision regarding final D&D plans, portions of the F Canyon facilities were used until 2008 to repackage noncompliant transuranic waste for shipment to the Waste Isolation Pilot Plant (WIPP) located in New Mexico (SRS F, 2008).

5.1.9 FB Line

FB Line, a 55,000 square foot area located in Building 221-F on top of F Canyon, was placed into operation in November 1954 to receive Pu-239 nitrate solution produced in F Canyon and convert it to a solid form (SRS at Fifty, 2002). Solutions were transferred from the canyon and concentrated in the FB Line. Then, in subsequent operations, the plutonium was precipitated, filtered, dried, and finally reduced to metal form. Process operations continued until March 2002, when all scheduled operations to stabilize plutonium-bearing materials from the SRS production era were completed. For about two years after those operations were completed, FB Line's focus was to stabilize and package legacy nuclear materials for safe, long-term storage. This process involved packaging materials using a process in which stabilized plutonium is placed in rugged, welded stainless steel cans. After materials were stabilized and packaged, they were shipped to other site locations until the Mixed Oxide Fuel Fabrication Facility was ready. The facility is now in safe shutdown status awaiting a DOE decision concerning D&D (SRS FB, 2007).

5.1.10 JB Line

The JB Line was a two-story plutonium-finishing facility constructed on top of the F Canyon building. Construction was started in 1956 but not completed until 1959 (Bebbington, 1990; WSRC-MS-2000-00061). The JB Line was built to permit high-capacity solvent extraction and plutonium finishing. The JB Line incorporated the trifluoride precipitation process and was built with facility safety improvements captured from the lessons learned in the original B Lines. Over following years, in-plant process and equipment improvements were made that resulted in JB Line production that exceeded government requirements for quantity and quality. The highest annual production for the JB Line was recorded in 1983 (WSRC-MS-2000-00061).

5.1.11 H Canyon

H Canyon is a 403,000 square foot facility originally constructed in the early 1950s to house fissile isotope separations process equipment. H Canyon is 835 feet long with several elevation levels to accommodate the various stages of material stabilization, including control rooms to monitor overall equipment and operating processes, and an equipment and piping gallery for solution transport, storage, and disposition. Operations involving the processing of depleted uranium fuel using the PUREX process were initiated in July 1955. Work in the canyon, including maintenance, was remotely performed by overhead bridge cranes so that worker exposure to radiation could be minimized. The thick, dense concrete walls that separate workers from the actual processing areas provide added protection.

The facility's operations also included chemical separations recovery of U-235 and Np-237 from aluminum-clad enriched uranium fuel tubes derived from SRS reactors and other domestic and foreign research reactors. In addition, H Canyon was equipped with capabilities to recover Np-237 and Pu-238 from special irradiated targets. Pu-238 was produced by irradiating recovered Np-237 in the SRS reactors. The Pu-238 was then recovered in H Canyon and used in power systems for the National Aeronautics and Space Administration's deep space exploration programs, such as the Cassini spacecraft that explored the planet Saturn.

For the historic H Canyon operations, nuclear materials (fuel tubes, oxides, etc.) were typically transferred from designated SRS storage areas to H Canyon, converted to solution, and then transferred through various process stages by which uranium, neptunium, and plutonium were separated. Occasional campaigns to separate U-233 and thorium were carried out in the 1960s. Waste material was transferred to the SRS high-level waste storage tanks for eventual vitrification in the Defense Waste Processing Facility.

In 1992, the H Canyon facilities were placed in standby status; however, there remained a significant inventory of highly-enriched uranium fuels and solutions in various components of the SRS process equipment. Between December 1995 and October 1997, DOE issued a series of decisions to resume H Canyon chemical separation operations in order to stabilize and safely manage most of the remaining SRS inventory of highly-enriched uranium (HEU). DOE also concluded that H Canyon could be used to support stabilization of Np-237 stored in H Canyon and a number of plutonium solids stored in F Area vaults.

In October 1997, H Canyon renewed operations to dissolve HEU into chemical solutions that could then be blended with natural uranium (NU) to form a source of low-enriched uranium (LEU) suitable for light water commercial nuclear power reactors. In July 2003, the first SRS LEU solution shipment was sent to an off-site facility operated by the Tennessee Valley Authority (TVA), which converted the LEU solutions into solid fuel for use in their power reactors at the Browns Ferry Nuclear Station. These LEU solution shipments continued through 2007.

Areas within H Canyon have been used for other purposes. For example, the H Canyon truck well was used to repackage the SRS inventory of radioactive transuranic (TRU) waste into containers suitable for relocation to DOE's Waste Isolation Pilot Plant (WIPP) in Nevada, which serves as DOE's long-term storage facility for all TRU waste materials (Washington Savannah River Company, 2008).

5.1.12 Old HB Line

The Old HB Line was a Pu-239 processing facility completed and brought online in 1953. It was periodically improved and upgraded throughout its operational life, and was shut down in 1984 when it was replaced by the New HB Line. It produced Pu-239 buttons from 1953 to 1960, and then was upgraded to support the NASA programs in the production of Pu-238 oxide primarily as a heat source for generating electricity for spacecraft going into deep space. The Old HB Line also processed neptunium oxide when SRS reactors were in operation.

Limited decontamination of the Old HB Line facility began in 1984 but was interrupted in 1986 due to funding restraints. At that time, the scrap recovery process equipment and the shielding for the neptunium oxide process equipment had been removed. The D&D efforts resumed in 1988. (Bebbington, 1990; DOE/EA-0948).

5.1.13 New HB Line

The New HB Line, is a 28,000 square foot plutonium/neptunium processing facility that was built in three phases during the early 1980s. The New HB Line plutonium processing facilities are located on top of the H-Area Canyon Building 221-H. The Frame Waste Recovery process is located within the 221-H building. The New HB Line facility also houses a vault for the storage of Pu-238 oxide product and scrap material.

There are three process lines. Phase I, or the Scrap Recovery Line, became operational in the late 1980s and is used to dissolve and dispose of legacy plutonium scrap. It is also used to dissolve legacy enriched uranium for blending into low-enriched uranium to be shipped to the Tennessee Valley Authority for fabrication into commercial power reactor fuel. The Phase I process converts solid nuclear materials into nitrate solutions and transfers those solutions to H Canyon for disposition. Phase II, or the Np-237/Pu-239 Oxide Line, can produce oxide (powder) material from Np-237 or Pu-239 nitrate solutions. Phase II started operations for the first time in November 2001. The plutonium material was shipped to FB Line for packaging in 3013 containers for long-term storage, and then to K Area for storage. The neptunium material is shipped to the Idaho National Laboratory for further processing and conversion to reactor targets for future Pu-238 production. Phase III, or the Plutonium-238 Oxide Production Line, was converted into a processing facility to open storage containers when necessary, and oxidize metals to allow for dissolution in the Phase I process area. Phase III supports missions to disposition legacy plutonium and uranium metals and oxides (DOE/EA-0948).

5.1.14 Naval Fuels Manufacturing Facility

The Naval Fuels Manufacturing Facility in Building 247-F was a two-story, 110,000 square foot enriched uranium fuel manufacturing facility built in the early 1980s. It operated from 1985 through 1989 to provide additional naval nuclear fuel manufacturing capacity for the cold war effort. The manufacturing processes employed a wide variety of acids, bases, and other hazardous materials. As the cold war wound down, the need for naval fuel declined. Consequently, the facility was shut down and underwent initial deactivation. All process systems were flushed with water and drained using the existing process drain valves. However, since these drains were not always installed at the lowest point in piping and equipment systems, a significant volume of liquid remained after initial deactivation was completed in 1990. At that time, a non-destructive assay of the process area indicated that as much as 34 kg of uranium might remain in equipment and piping. The 247-F Closure Project is currently underway that will result in the final D&D of the Naval Fuels Manufacturing Facility.

5.1.15 Building 235-F Plutonium Fuel Fabrication Facility (PUFF)

The 235-F Plutonium Fuel Fabrication Facility (PUFF), or Building 55, was a 55,000 square-foot operations facility containing Pu-238 hot cells. Operations to produce encapsulated Pu-238 oxide fuel forms and, on occasion, neptunium billets began in 1978. In December 1983, DOE completed Pu-238 fuel clad production for NASA's Galileo and Ulysses space missions at PUFF, after which the equipment was placed on operational standby status. Currently, the PUFF equipment is being used to provide safe storage of residual nuclear material resulting from past nuclear weapons production and other non-weapon uses that are no longer required.

5.1.16 Savannah River Technology Center

The Savannah River Laboratory research facilities include the main laboratory building, 773-A; an experimental physics laboratory, 777-M; the CMX and TNX process pilot facilities located near the 400-D Area; the Health Physics Laboratory, 735-A; and the equipment engineering laboratory and shops, 723-A. CMX was closed down in 1983 when its function (long-term flow testing of new fuel and target elements) was moved to 773-A. The 777-M operations were also discontinued in the 1980s when modern computers eliminated the need for many of the experimental measurements that were previously needed to support reactor charge design.

5.2 Radiological Exposure Sources from Savannah River Site Operations

ATTRIBUTION: Section 5.2 and its related subsections were completed by Ray Clark, Oak Ridge Associated Universities; Bryce Rich, Mel Chew and Associates, Inc.; Sam Chu, Mel Chew and Associates, Inc.; and Eugene Potter, Mel Chew and Associates, Inc. These conclusions were peer-reviewed by the individuals listed on the cover page. The rationales for all conclusions in this document are explained in the associated text

SRS operations involved several processes of the nuclear weapons development cycle (DOE, 1997; DNA-1251-1-EX) and played a significant role within the US nuclear weapons program. These processes include nuclear fuel fabrication; nuclear reactor operations; radiochemical separations; radionuclide production - including nuclear weapons materials; recycling uranium; refining, finishing and storing plutonium; and handling the associated radioactive waste. The following subsections provide an overview of the internal and external exposure sources associated with these operations for the class under evaluation.

5.2.1 Internal Radiological Exposure Sources from SRS Operations

At SRS, tritium produced most of the personnel exposure from internal deposition. Major sources of non-tritium intakes were potential uptakes of uranium and plutonium; secondary sources were mixed fission products and activation products.

Table 5-1 provides a summary of radionuclides that are recognized as potentially contributing to internal dose to SRS workers. These radionuclides of concern (ROCs) are listed in Chapter 5 of the *SRS Internal Dosimetry Technical Basis Manual* (WSRC-IM-90-139). These ROCs were identified by SRS as the most significant internal intake sources, or effective tracers for evaluating the significant dose-delivering radionuclides (ORAUT-TKBS-003, p. 64). The following subsections address these ROCs in turn.

Table 5-1: Potential Contributors to SRS Internal Dose

Radionuclide
H-3
Pu-238, Pu-239, Pu-240, Pu-241, Pu-242
U-233, U-234, U-235, U-236, U-238, and mixtures
Np-237
Am-241, Am-243
Cm-244
Natural thorium
Mn-54
Co-60
Zn-65
Sr-90 / Y-90
Ru-106
Sb-125
Cs-134, Cs-137
Ce-144
Eu-152, Eu-154
Tm-170
Cf-252 (including Cm-248 daughter)

5.2.2 External Radiological Exposure Sources from SRS Operations

SRS was built to produce the basic materials used in the fabrication of nuclear weapons, primarily tritium and Pu-239. Five reactors were built to produce nuclear materials (including nuclear weapons materials) through neutron irradiation of target materials in the production reactors. Support facilities included two chemical separations plants, a heavy water extraction plant, a nuclear fuel and target fabrication facility, and waste management facilities.

The fuel elements used in the reactors generated direct beta, photon, and neutron radiation fields, as well as the potential for surface and airborne contamination by uranium, TRU isotopes, and a wide variety of fission and activation products. Separations processes, research and development activities, and the collection and disposal of radioactive wastes also resulted in potential direct exposures to direct, beta, photon, and neutron radiation fields.

Neutron exposure sources consisted of: (1) the operating reactors; (2) the separations processes and product lines, which handled quantities of TRU isotopes and the associated spontaneous fission and α/n neutrons produced; (3) the analytical and production control laboratories to a lesser degree; (4) R&D facilities, which also handled TRU isotopes in significant quantities and associated neutron emissions; (5) vaults and TRU storage facilities; and (6) calibration facilities and others whose operation relied upon the use of neutron sources.

5.2.3 Incidents

NIOSH has found no evidence or documentation of any incidents that would have resulted in very high exposures (similar to those received in a nuclear criticality accident).

Starting in March 1954, some incidents involving work with radioactive materials were documented, and worker exposures were assessed when incidents involved radioactive sources (SRS Dosimetry, 1959; SRS Dosimetry, 1962; SRS Dosimetry, 1974a; SRS Dosimetry, 1974b; SRS Dosimetry, 1989; DPSP-55-454-2). Reports of these incidents were retained in SRS Dosimetry Special Hazards Investigations files. Copies of many of these reports have been obtained through 1989 and are in the SRDB. SRS Special Hazards bulletins required Construction supervision to be part of incidents involving radioactive materials and construction workers (DPSOP-40, pdf pp. 71, 183, 248).

NIOSH completed a search of the SRS Site Incident database that lists 332 exposure incidents from 1954 through 1996. The search found that 481 worker incident records with "urine results." Further in-depth investigation revealed 31 workers received confirmed intakes of radioactivity. Also listed in the incident reports were 12 neutron exposures and 12 external exposures (Singh, 2007). NIOSH found incident monitoring data in NOCTS for former SRS workers who were identified in incident records.

Construction workers were involved in some of the recorded incidents beginning in 1954. In the first 100 incidents from the SRS Dosimetry Special Hazards Investigations files (March 1954 – March 1959), construction workers were involved in twenty-six incidents. In each incident, workers were monitored for contamination. Depending upon the type of incident, they were also monitored for external dose and for internal uptake of radioactivity.

6.0 Pedigree of Savannah River Site Data

This section answers key questions prior to performing a feasibility evaluation. Data Pedigree addresses the background, history, and origin of the data. It requires looking at site methodologies that may have changed over time; primary versus secondary data sources and whether they match; and whether data are internally consistent. All these issues form the bedrock of the researcher's confidence and later conclusions about the data's quality, credibility, reliability, representativeness, and sufficiency for determining the feasibility of dose reconstruction. The feasibility evaluation presupposes that data pedigree issues have been settled.

Historical information gathered for SRS indicates that the management of individual radiation exposure records began at hiring with the accumulation of data pertinent to the individual and any previous occupational radiation exposure. Appropriate radiation monitoring badges or devices were issued and accountability was established (DPSPU-75-30-7). Site dosimetry records have been maintained by the same organization since the site's inception. Employee identification numbers were used as the sole tracking method for the early years. In the early 1960s, the practice of re-using the identification numbers of terminated employees was begun, which complicated the unique tracking of an employee's historical exposure. Social security numbers were included in the dose records in 1973. From 1951 to 1957, personnel exposure was recorded on various data forms and stored in individual file folders. From 1958 through 1972, employee exposure information was sent to a central computing facility for entry into a computer file. Quarterly computer-generated compilations of exposures for the cycle, quarter, year, and site were maintained in "logbooks."

In 1973, a more sophisticated computer program, HP Master File, replaced the previous system. Each year's records were stored on a single magnetic tape. The HP Master File also produced bioassay sampling schedules, whole body/chest count schedules, and tritium sample results for monthly dose equivalent calculation as well as maintaining accountability for film badges.

In 1979, a computerized system to produce annual dose reports to employees ("report cards") was initiated. The Health Protection Annual Radiation Exposure History (HPAREH) system was developed by building and verifying a history file of annual radiation exposure data from handwritten data, logbooks, and magnetic tape. The first report cards were issued in 1980 (for 1979), and the HPAREH database was updated from the HP Master file for each subsequent year.

In 1989, the HP Master File system was replaced by the Health Protection Radiation Exposure Database (HPRED) system. Previously, data were either hand-entered on computer input sheets or processed on magnetic tapes and then sent to the site's computer group. With HPRED, data was input either using terminals or directly from the TLD badge reader system. In 2004, HPRED was replaced with a commercial product (ProRad). From 1951 to 1983, visitor exposure records were posted manually on individual 3-inch by 5-inch cards and filed by year in a card file. The cards were separated into plant or construction visitors and placed in alphabetical order (WSRC-RP-95-234).

6.1 Internal Monitoring Data Pedigree Review

ATTRIBUTION: Section 6.1 and its related subsections were completed by Sam Chu, Mel Chew and Associates, Inc.; and Eugene Potter, Mel Chew and Associates, Inc. These conclusions were peer-reviewed by the individuals listed on the cover page. The rationales for all conclusions in this document are explained in the associated text.

NIOSH has found that SRS policies for the collection and maintenance of employee monitoring data are sufficient for dose reconstruction in terms of the pedigree parameters described above. Upon request, the SRS provides database printouts and copies of original hardcopy bioassay and whole body count/chest count records. For most of the site's history, the data are provided as originally recorded, but since 1990, excreta analysis records have been maintained in an electronic database and computer-generated reports are provided.

Since no electronic data are available before 1990, data from NOCTS have been entered into a database for the purpose of creating a co-worker model for workers who were unmonitored or for whom records are unavailable. There are approximately 382,000 *in vitro* bioassay records available for analysis. After entry, these data were subject to 100% verification by a second person. NIOSH has access to logbooks containing bioassay starting in 1954. A comparison of the computerized printout sheets and logbook entries for a number of claimants was completed to establish credibility and consistency of the internal dosimetry data. No evidence of systematic errors or significant differences between the printouts and hardcopy data was observed.

NIOSH reviewed entries in four bioassay logbooks covering a period of six years as a representative sample of the data. Of the 200 logbook entries reviewed, 62 were claimants in the NOCTS database. Three claims contained no data corresponding with the logbook entries (< 5%). Fifty-seven claims had corresponding data (92%). Forty-two percent of the claimants were in construction-related positions.

6.2 External Monitoring Data Pedigree Review

ATTRIBUTION*: Section 6.2 and its related subsections were completed by Bryce Rich, Mel Chew and Associates, Inc.; Eugene Potter, Mel Chew and Associates, Inc; and Darin Hekkala, Dade-Moeller, Inc. These conclusions were peer-reviewed by the individuals listed on the cover page. The rationales for all conclusions in this document are explained in the associated text.*

Records of radiation dose to individual workers wearing personnel dosimeters are available for SRS operations beginning in 1951. Dose from these dosimeters was recorded at the time of measurement and routinely reviewed by SRS operations and radiation safety staff for compliance with radiation control limits. Based on the SRS documentation, there does not appear to be any significant administrative practice that would affect the integrity of the recorded doses for SRS workers with the exception that some workers exposed to natural uranium might not have been monitored prior to July 1955.

For monitored workers, NIOSH has obtained the complete set of external dosimetry records on the quarterly reports since 1958, including the visitor cards. Although this is the complete dataset, the primary source of the co-worker dose model in ORAUT-OTIB-0032 was the Health Protection Annual Radiation Exposure History Database (HPAREH) developed in 1979. This database consists of annual doses to workers (shallow, deep, neutron, and tritium) and was used from 1980 to 1989 to produce the annual reports. Some workers who terminated employment prior to 1979 and visitors are not included in HPAREH. However, external exposure records for these employees are available on paper, either in the employee's individual dosimetry file or in visitor records. Table 6-1 presents the number of annual dose records contained in the HPAREH database for SRS workers along with the total number of monitored workers listed in WSRC-RP-95-234.

In April 1989, the Health Protection Radiation Exposure Database (HPRED) was implemented. This system interfaced directly with the TLD badge reader system. In 1990, visitor records were included in this database.

In 2004, HPRED was replaced by a commercial access control and radiological records management system, the ProRad database. Advantages of the system over the previous one included: real-time availability of individual actual and estimated dose totals, automated personnel qualification checks, improved efficiency and real-time availability of radiological information (SRS Intake, 2004).

Table 6-1: Summary of Available Monitoring Data for SRS Workers					
	No. of Monitored Workers per WSRC-RP-95-234	Total Number of Annual External Dose Records in HPAREH			
Year		No. of Monitored Workers	Shallow Dose Records	Deep Dose Records	Neutron Dose Records
1950		2	0	0	0
1951	50	29	0	0	0
1952	400	270	177	177	0
1953	1700	992	594	594	1
1954	4800	2736	1391	1391	1
1955	5500	3177	2012	2012	1
1956	5600	3235	2740	2740	10
1957	5900	3375	2971	2971	19
1958	6200	3500	3132	3132	17
1959	5756	3189	2804	2804	7
1960	5670	3425	3253	3253	17
1961	5819	3537	3358	3358	10
1962	6332	3630	3474	3474	17
1963	6219	3815	3501	3501	33
1964	5498	3814	3372	3372	120
1965	4977	3682	3468	3468	178
1966	5032	3720	3329	3329	184
1967	5042	3819	3603	3603	223
1968	4875	3822	3670	3670	365
1969	4705	3797	3671	3671	517
1970	5224	3810	3412	3412	484
1971	4856	3956	3302	3302	597
1972	4737	4175	3384	3384	531
1973	5639	4369	3534	3534	462
1974	5653	4472	3809	3809	480
1975	6129	5056	4329	4329	424
1976	6336	5456	4784	4784	486
1977	6318	6164	5057	5057	517
1978	7647	7067	5760	5760	403
1979	7615	7763	6286	6286	502
1980	8066	8213	6571	6571	675
1981	8951	9131	6764	6764	840
1982	9192	9371	6515	6515	608
1983	10195	10396	7393	7393	670
1984	11740	12006	7074	7074	781
1985	13950	14253	8323	8323	935
1986	14236	14544	10949	10949	802
1987	14096	14565	11700	11700	889
1988	15458	15920	12438	12438	787
1989	17468	17925	13674	13674	733
1990	20535	20934	11350	11350	697
1991	20428	20832	19908	19908	2135

Table 6-1: Summary of Available Monitoring Data for SRS Workers					
	No. of Monitored Workers per WSRC-RP-95-234	Total Number of Annual External Dose Records in HPAREH			
Year		No. of Monitored Workers	Shallow Dose Records	Deep Dose Records	Neutron Dose Records
1992	19320	19682	18734	18734	2083
1993	16747	17078	16077	16077	2926
1994	13983	14120	12587	12587	3057
1995	(see Note)	13923	12450	12450	3385
1996	N/A	12716	11400	11400	3414
1997	N/A	12293	11069	11069	3382
1998	N/A	11620	10484	10484	3261
1999	N/A	11150	10095	10095	3306

Note: Data values in Column 2 end at 1994 because WSRC-RP-95-234 was published in 1995.

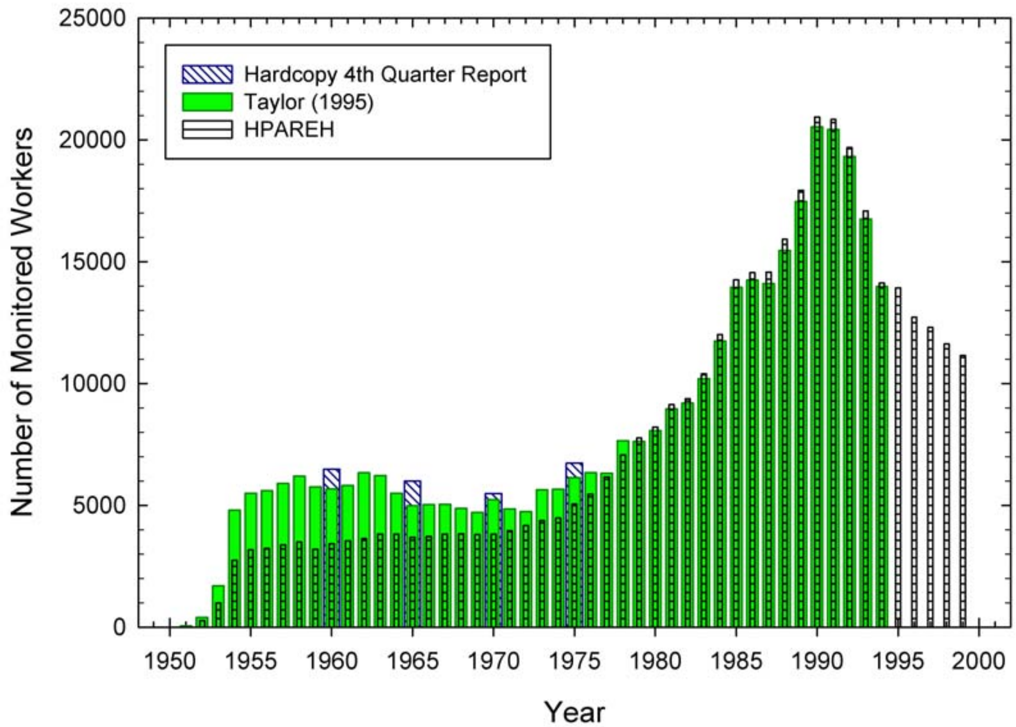


Figure 6-1: Comparison of the Number of Monitored Workers in HPAREH vs. WSRC-RP-95-234 and Hardcopy 4th Quarter Reports Obtained from SRS in August 2008

As mentioned above, not all historical monitoring data was stored in the HPAREH database. The data for some workers who terminated employment prior to 1979 and visitors are not included in HPAREH since these data was not needed to produce the annual reports. Thus for pre-1979 data, a comparison of the number of workers in the hardcopy quarterly reports with HPAREH and the data in WSRC-RP-95-234 was conducted. As illustrated in Figure 6-1, for 1960 approximately 53% of the monitored workers are in HPAREH. The percentage of workers included in HPAREH generally increased with time till 1979 when all workers were included.

The petitioners directly questioned the ability to reconstruct doses in light of this missing data in HPAREH. For this evaluation, a review of the annual dose distributions was conducted in five-year increments between 1960 and 1975 (Figure 6-2). Based on this evaluation, the annual dose distributions using all of the monitoring data in the 4th Quarter reports were less than the dose distributions in HPAREH. As a result, even though HPAREH does not contain all of the monitoring data, the annual dose values used in ORAU-OTIB-0032 may be used to bound dose.

The lower border of each bar is the 25th percentile of the distribution, the solid line midway in each bar represents the median, the dashed line in each bar is the mean of the distribution, the upper border of each bar is the 75th percentile, the upper whisker is the 90th percentile, and the circle above each bar is the 95th percentile of the distribution.

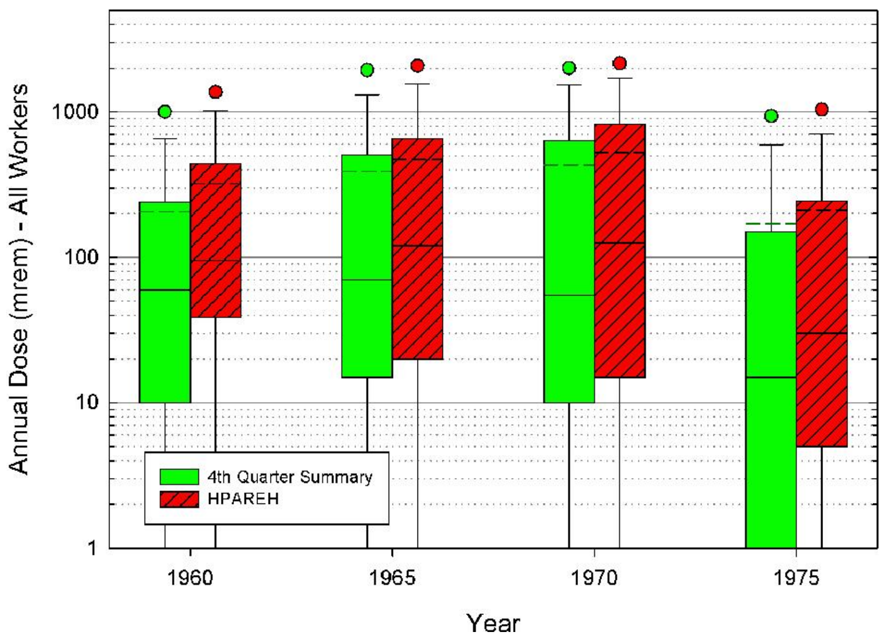


Figure 6-2: Comparison of Annual Dose Distributions for Construction Trades Workers in HPAREH and the 4th Quarter Reports

A subsequent comparison was conducted evaluating only the construction worker data and the 4th Quarter Reports. Table 6-2 presents the total number of construction trades workers in HPAREH and the 4th Quarter Reports. In general, less than half of the Construction Trades workers were listed in HPAREH compared to the 4th Quarter Reports. As with the all worker comparisons, HPAREH annual dose distribution was generally higher than the 4th Quarter Reports, as shown in Figure 6-3. The exception was 1970 in which the 4th Quarter Reports indicated a slightly higher 75th percentile.

Table 6-2: Comparison of Construction Trades Workers to HPAREH

Year	Number of Monitored Construction Trades Workers	
	HPAREH	4th Quarter Reports
1960	202	747
1965	236	531
1970	211	312
1975	568	1011

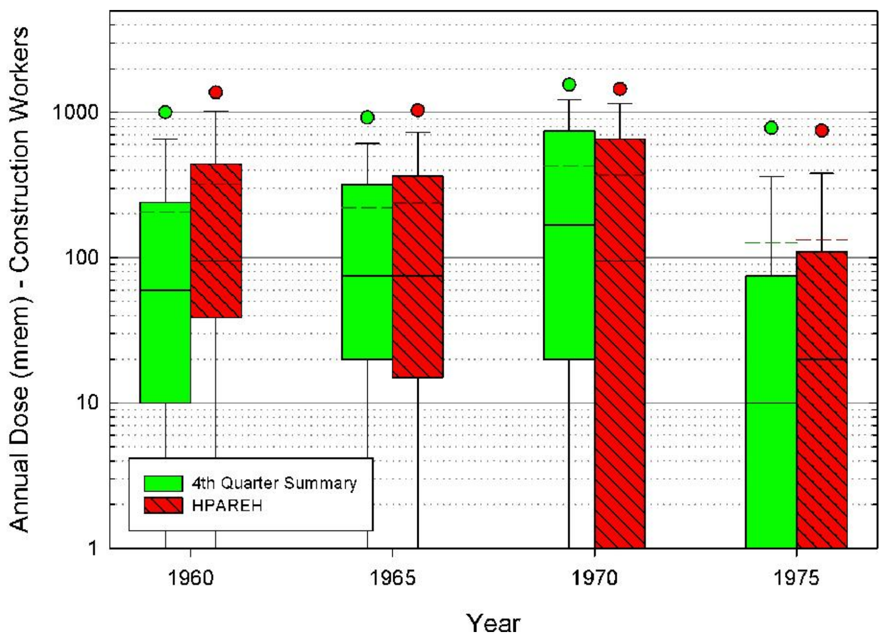


Figure 6-3: Comparison of Annual Dose Distributions for Construction Trades Workers in HPAREH and the 4th Quarter Reports

The nature of construction work is periodic; construction workers would enter and leave before the annual reports were produced. To further evaluate the effect of this periodic work on the annual dose distributions, a complete roster of all construction workers was obtained for 1960. While HPAREH only listed 202 Construction Trades Workers, a total of 747 were listed on the 4th Quarter Annual Report. When the annual roll was reviewed which combined the 1st, 2nd, 3rd, and 4th quarters, a total of 1399 Individual Construction Trades Workers were identified, nearly double the 4th quarter results and 7 times the number in HPAREH. The 1960 annual doses for these workers were coded and compared (see Figure 6-4). As with the other distributions, as more workers were included, the annual dose distributions decreased. The likely cause of this bias is that more of the experienced crafts were possibly employed for a longer period of time and used for more of the higher exposure jobs. This effectively resulted in a slight claimant favorable bias in the HPAREH database.

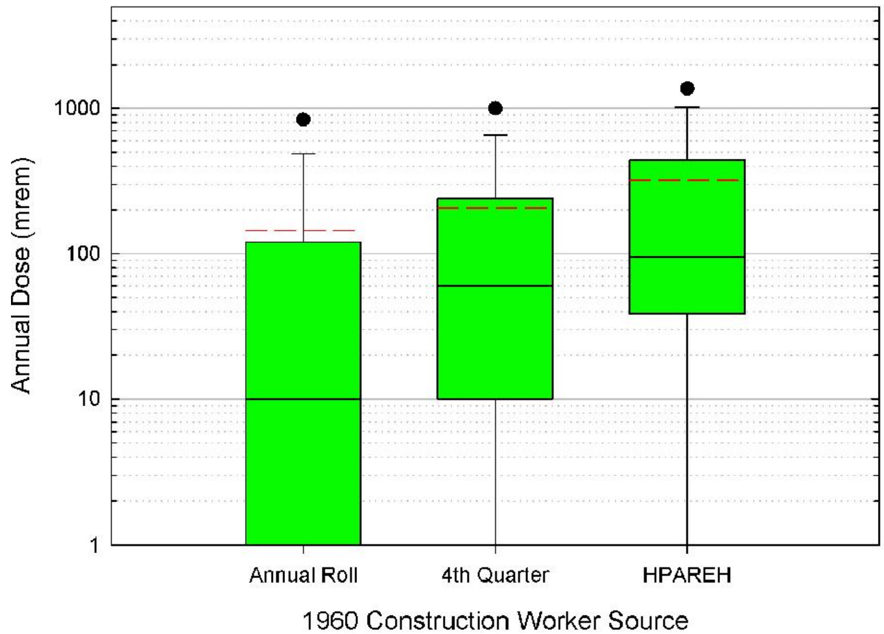


Figure 6-4: Comparison of 1960 Annual Dose Distributions Using the Annual Roll, 4th Quarter Reports, and HPAREH

Based on the review of the annual dose distributions, and specific evaluation of the construction trades workers, NIOSH concludes that the HPAREH database used in ORAU-OTIB-0032 is sufficiently robust. For the purpose of this evaluation, this pedigree assessment supports the use of these external monitoring data for determining the feasibility of bounding external dose for the class under evaluation when ORAUT-OTIB-0052 guidance is applied to the co-worker model.

Neutron Dose

Since 1972, the SRS has used Thermoluminescent Neutron Dosimeters (TLNDs) to measure personnel neutron doses at SRS. Since TLNDs are recognized as state of the art dosimetry for neutrons, neutron doses in HPAREH post 1972 are considered sufficiently robust for the development of a co-worker model. Thus, further evaluation of the pedigree of these measurements is not warranted.

However, prior to 1972 the neutron doses are significantly limited due to the use of NTA film. In order to use this NTA data, correction factors must be applied to correct for under-response due to energy limitations. Based on a review of SRS reports recently captured as part of this SEC evaluation, SRS conducted direct measurements comparing the new TLND, area survey measurements and NTA film (DP-MS-69-004, 1969). Table 6-3 provides the direct work place comparison at the SRS Plutonium Finishing Area. The average under-response of the NTA film was approximately a factor of four.

Table 6-3: Comparison of Bonner Sphere, TLND, and NTA Film Measurements at the SRS Plutonium Finishing Area				
Location	**12-in Bonner Sphere (mrem)**	**TLND (mrem)**	**NTA* (mrem)**	**Correction Factor**
Plutonium Finishing Area	80	89	31	2.6
			26	3.1
			14	5.7
		90	19	4.2
			19	4.2
			14	5.7
Average	80	89.5	20.5	3.9

* The NTA Film was calibrated using a high energy PuBe source.

As noted in the SRS TBD, personnel monitoring for neutrons was only required when workers entered a radiation area where the neutron dose rate exceeded 1 mrem/hr. Thus, some unmonitored exposure to neutrons was likely. This is the primary reason the TBD uses a neutron-to-photon ratio (N:P) to assess the unmonitored neutron dose.

From an external dose pedigree standpoint, validation of the neutron-to-photon ratio listed in the TBD will require significant work (as was required for the Hanford SEC evaluation). This work will be conducted after this report is submitted. Currently, the complete set of personnel neutron dose measurements has been reviewed on site by the SEC evaluation team and has subsequently been requested from SRS. As of this writing, these data are currently undergoing classification review and have not been formally evaluated.

During the on-site review of the NTA data, NIOSH noted that prior to 1961 there appears to be much less NTA data available for verification of the N:P ratio. Due to this lack of data, NIOSH also reviewed Radiation Survey Reports to determine if workplace neutron measurements were conducted in parallel with photon survey measurements. NIOSH has verified that this in fact occurred. The paired measurements have been recorded on individual Radiation Survey Log Sheets (RSLS) in the SRS records holdings. The combination of the corrected NTA data along with the survey data will be used to verify the N:P ratio in the SRS TBD. This methodology is discussed in more detail in Section 7.2.

7.0 Feasibility of Dose Reconstruction for the Class Evaluated by NIOSH

The feasibility determination for the class of employees under evaluation in this report is governed by both EEOICPA and 42 C.F.R. § 83.13(c)(1). Under that Act and rule, NIOSH must establish whether or not it has access to sufficient information either to estimate the maximum radiation dose for every type of cancer for which radiation doses are reconstructed that could have been incurred under plausible circumstances by any member of the class, or to estimate the radiation doses to members of the class more precisely than a maximum dose estimate. If NIOSH has access to sufficient information for either case, NIOSH would then determine that it would be feasible to conduct dose reconstructions.

In determining feasibility, NIOSH begins by evaluating whether current or completed NIOSH dose reconstructions demonstrate the feasibility of estimating with sufficient accuracy the potential radiation exposures of the class. If the conclusion is one of infeasibility, NIOSH systematically evaluates the sufficiency of different types of monitoring data, process and source or source term data, which together or individually might assure that NIOSH can bound either the maximum doses that members of the class might have incurred, or more precise quantities that reflect the variability of exposures experienced by groups or individual members of the class as summarized in Section 7.6. This approach is discussed in OCAS's SEC Petition Evaluation Internal Procedures which are available at http://www.cdc.gov/niosh/ocas. The next three major subsections of this Evaluation Report examine:

- The feasibility of reconstructing internal radiation doses. (Section 7.1)
- The feasibility of reconstructing external radiation doses. (Section 7.2)
- The bases for petition SEC-00103 as submitted by the petitioner. (Section 7.3)

Members of the proposed SEC Class were DuPont employees plus members of private contractor organizations and were provided identical Health and Safety coverage as were other operations and maintenance workers. Health Physics support of Construction activities began in 1953. Most workers were provided dosimetry, protective clothing and equipment, and needed bioassay coverage when they entered a radiologically-controlled work area. Health Physics services provided to construction workers are discussed in DPSP-55-454-3 and DPSOP-40. By procedure, all radiation and contamination control procedures applied to "All departments and Construction" (DPSOP-40).

7.1 Evaluation of Bounding Internal Radiation Doses at Savannah River Site

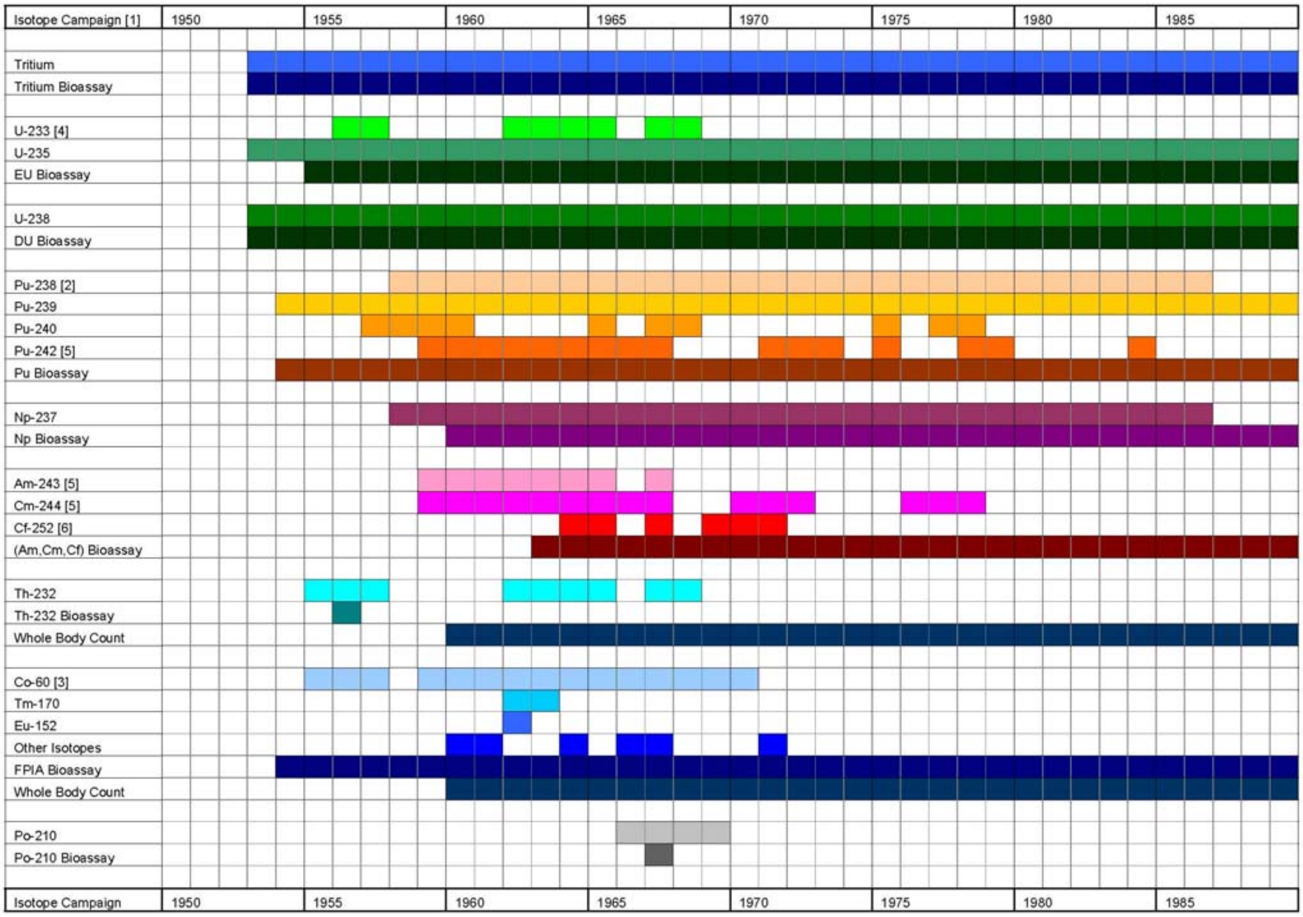
ATTRIBUTION: Section 7.1 and its related subsections were completed by Mike Mahathy, Oak Ridge Associated Universities; Eugene Potter, Mel Chew and Associates, Inc; Sam Chu, Mel Chew and Associates, Inc.; Robert Morris, Mel Chew and Associates, Inc.; and Will McCabe, MJW Corporation. These conclusions were peer-reviewed by the individuals listed on the cover page. The rationales for all conclusions in this document are explained in the associated text.

To evaluate the petitioner's concerns regarding internal exposures to special radionuclides produced at SRS, NIOSH investigated the source term, available personnel monitoring records, and air sample data, as needed. Based on this information, NIOSH made a determination as the feasibility of reconstructing internal doses for each radionuclide.

The principal source of internal radiation intakes for members of the class under evaluation was tritium. Major sources of non-tritium intakes were potential uptakes of uranium and plutonium; secondary sources were mixed fission products, activation products, and thorium (ORAUT-TKBS-0003).

The majority of SRS workers was monitored for internal intake of radionuclides and has internal monitoring data. This point is reflected in the NOCTS claimant files for which 1467 of the 1798 construction worker claim files contained internal monitoring data. However, the possibility exists that some workers who should have been monitored were not, or that the data for some workers who were monitored has been misplaced. As a result, NIOSH is developing a co-worker model based on the claimant data in NOCTS. There are approximately 382,000 *in vitro* bioassay records available for analysis in NOCTS. A separate study has been completed that established the principle that, under certain conditions, the data in NOCTS are representative of a site's population generally (ORAUT-OTIB-0075). All of the SRS NOCTS data has been entered into a database, and the co-worker model is in preparation. In addition, ORAUT-OTIB-0052 indicated that construction trades workers had more plutonium bioassay measurements below the reporting limit compared non-construction workers. ORAUT-OTIB-0052 also found that for the positive bioassay, the non-construction workers results were generally higher than construction trades workers.

A comparison of isotope production at SRS versus internal monitoring for intakes of those isotopes is presented in Figure 7-1 and Table 7-1. Note the bioassay start dates in Figure 7-1 correspond to the start dates of bioassay in NOCTS and not necessarily the site.



[1] Source: *Savannah River Site at Fifty* (SRS at Fifty, 2002, p. 453), table entitled *Isotope Production at SRS over Time*.

[2] This timeline (black bar) includes the first production of an isotopic heat source at SRS in 1958. This program continued until 1986 (pp. 357, 429).

[3] This timeline includes high-activity Co-60 generation for isotopic heat sources that began in May 1955 (pp. 356, 432).

[4] This timeline includes thorium irradiation as part of Eisenhower's Atoms for Peace program with the first production of U-233 in 1955-56 (p. 350)

[5] This timeline includes the Curium I and II, High Flux, and Transplutonium I and II programs that produced Pu-242, Am-243, and Cm-244, including Cf-252, Es-255, Fm-257, Fm-258, and Co-60 from the High Flux Program (pp. 357, 432, 433).

[6] This timeline includes the Californium I program that began in August 1969 and continued until November 1970 and produced milligrams of Cf-252 (p. 433).

Figure 7-1: SRS Isotope Production vs. Internal Monitoring for Intakes

Table 7-1: Bioassay Start Dates for Specific Isotopes and Suitable Bioassay Technique					
Production Isotopes	**Bioassay start**	**Logbook Ref. Box**	**SRS ID TBD Rev. 8, 2001**	**SRS ID TBD Page**	**Bioassay Technique**
Thulium-170	1954 (FP)	M270-4639-3			beta emitter, 84 keV gamma @ 3%. Gross Beta
Europium-152	1960 (WBC)		*in vivo*	5-12	whole body count or chest count
Californium-252	1965 (TKBS)		urinalysis or *in vivo*	5-3	gross alpha or chest count
Polonium-210	2/7/1967	QH 211-1217-94-006			gross alpha
Special programs, other isotopes	2/16/1961	QH 211-1217-94-006			Isotope specific
Plutonium-242	Gross alpha				gross alpha
Curium-244	11/19/1963	QH 211-1217-94-005	urinalysis or *in vivo*	5-3	gross alpha or chest count
Americium-243	1956	QH 211-1217-94-006		5-4	gross alpha
Plutonium-238	4/5/1954	QH 211 0886-93-015	urinalysis or *in vivo*	5-5	gross alpha, alpha spectroscopy, chest count
Plutonium-240	4/5/1954	QH 211 0886-93-015	urinalysis or *in vivo*	5-6	gross alpha, alpha spectroscopy
Cobalt-60 (IA)	5/1956	QH 211-1217-94-006	*in vivo*	5-15	whole body count
Uranium-233	Gross alpha				gross alpha
Tritium	7/29/1953	M270-4639-3	urinalysis	5-15	gross Beta
Plutonium-239	4/5/1954	QH 211 0886-93-015	urinalysis or *in vivo*	5-6	gross alpha, alpha spectroscopy
Neptunium-237	12/6/1953	QH 211-1217-94-006	urinalysis or *in vivo*		gross alpha, whole body count or chest count
Thorium-232			urinalysis or *in vivo*		

NIOSH has obtained the hardcopy SRS bioassay logbooks dating back to 1954. Since 1990, excreta analysis records have been maintained in an electronic database. The database identifies baseline (new hire), routine, special, and termination measurements. Prior to the electronic database, excreta results were kept on employee bioassay cards. Chemical separation and alpha counting were generally used prior to 1994. NIOSH has obtained alpha analysis procedures for determination of plutonium, uranium, and neptunium (DPSOP-47); polonium alpha counting; and thorium alpha counting. The urinalysis for trivalent actinides (americium, curium, and californium), dating back at least to the mid-1960s, consisted of directly plating the sample and gross alpha counting. The plutonium urinalysis program began in 1954. Approximately 200 samples were analyzed for thorium in 1956. Chemical separation and alpha spectroscopy have been used since 1994 to identify specific isotopes (e.g., U-234, U-235, and U-238).

Tritium urinalyses were available from start-up, but increased when SRS implemented liquid scintillation counting in 1958 (Health Physics Aspects, 1958). Tritium results in the 1990s were listed on the same summary form as external dose monitoring results.

Uranium urinalyses were also available from start-up. SRS technical documentation indicates that for earlier monitoring periods, the designations "enriched" and "depleted" reported for uranium analysis 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. Activity fractions for depleted, recycled, and highly-enriched uranium are listed in the TBS. Neptunium analyses were implemented by 1959. (ORAUT-TKBS-0003, pp. 65-71) Thorium urinalyses were performed starting in 1956.

Intakes of gamma-emitting fission products were monitored by urinalysis, beginning in 1954 when the reactors went on line. The start date for urinalysis for strontium radioisotopes was December 1954, the month following the first separations work in F Canyon (Bioassay Logbook, 1954-57). However, Boni discusses a bioassay procedure in use at the time for rapid estimation of the total amount activity of beta-gamma-emitting isotopes in a urine specimen with recoveries of Sr-89, Sr-90/Y-90, Zr-95/Ni-95, Ce-144/Pr-144, Fe-59, Cr-51, and Zn-65 greater than 90% and Co-60 of 85% (DPSPU-58-030-011).

Whole-body counting has been performed since 1960 when the SRS Whole Body Counting Facility was completed. The first chest counting was with sodium iodide (NaI) detectors in 1965. Chest counting for low-energy photons was improved in the early 1970s using phoswich detectors. *In vivo* count records are not kept in a central electronic database; each count is maintained as a single hardcopy record. A whole-body count record and a chest-count record on the same day are listed on a single hardcopy record (ORAUT-TKBS-0003, p. 71).

Air sample records were considered workplace monitoring records as opposed to intake monitoring records, and hence, 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 (ORAUT-TKBS-0003, p. 75). The air sample results were recorded on Air Sample Log Sheets (ASLS). NIOSH has located several hundred boxes of the records through a search of the SRS EDWS records system.

Historically, SRS had an extensive radiation safety monitoring program to measure radiation exposure in the workplace, including contamination surveys (ORAUT-TKBS-0003, p. 89). Radiation surveys are recorded on the Radiation Survey Log Sheets (RSLS). The RSLS are filed by facility and time. These records are sequentially numbered so it can be determined with certainty if some are missing. However, like air monitoring data, these records are not likely to be found in the individual employee's records.

7.1.1 Radionuclides of Internal Concern

Each radionuclide of internal concern is discussed in the following subsections. A discussion of personnel monitoring and a determination of feasibility of dose reconstruction is presented for each.

7.1.1.1 Tritium

Tritium has been one of the main production materials throughout the life of the Savannah River reactors. Historically, tritium resulted in personnel intakes because of the large quantities present at SRS. Tritium does not occur in abundance in nature but can be produced by bombarding Li-6 targets with neutrons to create the unstable Li-7 isotopes which decay to become tritium through alpha emission within the reactors. Tritium-containing irradiated targets were then transported to the 200-F Area initially and the 200-H Area Tritium Facility for purification and packaging into usable form. The production of tritium began in 1953 and continues to the present time. Between October 1955 and 1964, SRS produced the majority of the tritium in the US. SRS has produced many kilograms of tritium during its production period (SRS at Fifty, 2002).

The first production reactor started in December 1953. Tritium concentrations in the moderator steadily increased with reactor operation. An April 1954 HP Area Survey Report stated for the 100R Area "The P-10 content of the process water has reached a level such that breaking of process water lines requires a Scott Air Pak." (Health Physics, 1954) A July 24, 1954 report identifies that a working level = 1E-5 uc/cc for tritium and that the "only positive bioassay results obtained for tritium to date are for two men who cleaned up a spill in E Plant… Bioassay results were above the sensitivity limit but well below the permissible body burden." (Summary of Tritium, 1954) The reported average tritium concentration in the five reactors was 1,000 μCi/mL in January 1956, 2,000 μCi/mL in January 1957, and 3,000 μCi/mL in December 1957. The number of tritium air samples >1E-5 μCi/mL (requiring job restrictions such as limiting stay time and or personal protective equipment) was 14 in January 1956, 173 in January 1957, and 217 in December 1957. There were four significant tritium bioassay results (>1 mCi assimilation) in 1956 and 49 in 1957 (Health Physics Aspects, 1958). There were no significant bioassay results prior to 1956 (DPSPU-58-030-014).

As of August 26, 1957 all personnel were subject to requests for bioassay samples by Health Physics, as dictated by results of the survey program. Personnel subjected to air activity above prescribed levels were required to submit samples as requested by Health Physics. In addition to the above, all construction personnel who worked during shutdowns were to submit bioassay samples, as follows:

a. At the end of the first week, welders and grinders, pipefitters and/or other personnel who have worked in close proximity to open water lines, for tritium urinalysis.

b. Welders and grinders at completion of their work in the area, but prior to leaving the area, for induced activities (750 mL). NOTE: Sample should be started soon enough for completion prior to leaving the area (e.g., approximately ten days before).

c. Personnel at completion of shutdown or upon completion of work in the area, but prior to leaving the area, for tritium analysis (Health Physics, 1957).

Tracking and follow-up procedures were in place to obtain samples from workers who did not submit a sample, as evidenced by a memo to M.W. Hartnett requesting 750 mL bioassay samples from over 100 Pipe and Boilermaker crafts personnel who failed to provide samples following the 100-K Area shutdown (Bio-Assay Samples, 1957).

Tritium was recovered in the tritium processing facilities (200 Areas). 232-F was built as an interim facility in 1953 – 1954. Tritium extraction operations began in October 1955 after the reactor and separations start-ups. In July 1957, a larger tritium facility began operation in 232-H. In August 1958, the capacity of 232-H was doubled. Also, in 1957 a new task was assigned to SRS – the loading of tritium into "reservoirs" that would be actual components of thermonuclear weapons. In October 1958, the F-Area Tritium Facility was shut down permanently (WSRC-IM-94-39).

In addition to tritium exposures during the tritium recovery process in the 200-H Area facilities, there was the potential for tritium exposure to workers in the reactor buildings from leaks from the primary system. The reactors used "heavy water" (high in the deuterium isotope of hydrogen) as the moderator, which resulted in the production of tritium by neutron absorption in the deuterium isotope. Thus, there was the potential for tritium exposure to workers in the reactor buildings from leaks from the primary system. Construction workers would have been exposed during maintenance operations.

Personnel Monitoring for Tritium

NIOSH has obtained tritium bioassay data for workers starting in 1953. In 1979, the Health Protection Annual Radiation Exposure History Database (HPAREH) was developed, which recorded annual doses to workers (shallow, deep, neutron, and tritium). Information regarding the reporting levels and detection limits at SRS is included in the SRS TBD. Information in the bioassay logbooks show the highest pre-HPAREH results for the applicable monitoring years.

Airborne Monitoring for Tritium

Tritium exposure was controlled through a combination of facility design, training, personnel protective equipment, detection and measurement, and work controls (DPSPU-58-030-014; DPSPU-62-30-5; DPSPU-62-030-025A). Tritium air concentrations were continuously monitored by stack monitors and by portable instruments. Three instruments were typically used because they met most requirements: a Kanne chamber (WSRC-IM-94-39), a portable tritium monitor ("sniffer"), and a small ionization chamber for taking grab samples.

Controls for tritium in air are interrelated with bioassay and annual dose limits. For control purposes, a 20 μCi/L value has been used; personnel who assimilate tritium above this level are removed from all work where further uptakes of tritium may occur (DPSPU-62-030-025A). When the assimilated tritium exceeds 40 μCi/L, the employee is removed from all work related to radiation. In either event, the employee cannot return for further work in tritium atmospheres until the body burden is 10 μCi/L or below. Good control has been realized by requiring personnel working in tritiated atmospheres to wear plastic suits for line breaks or when airborne concentration exceeds 20E-5 μCi/cc.

Table 7-2 shows the tritium exposure experience at SRS for the years 1954 through 1961.

Table 7-2: SRS Tritium Exposure Experience

Year	1954	1955	1956	1957	1958	1959	1960	1961
No. Assayed	2190	2310	2230	4000	3600	2590	2720	2690
Tritium Assimilations								
0.37 - 1.0 millicurie	*	*	*	*	307	156	452	401
>1.0 millicurie	0	2	13	49	33	22	79	73
Max. Assimilation, millicurie	-	1.2	153	8.8	3.6	3.2	3.9	14.6

Source: DPSPU 62-30-5
* Data not tabulated prior to 1958.

Feasibility of Bounding Tritium Doses

Construction workers exposed to tritium were monitored by urinalysis; therefore, personal bioassay data are available and reported by the site in their individual monitoring records. These data can be used to bound the individual worker's dose from tritium. For workers who were not monitored, or whose records provided by the Department of Energy may be incomplete (e.g., missing data), tritium dose may be assigned based on information in Section 4.4.3 of the TBD (ORAUT-TKBS-0003). That is, based on their job category (ORAUT-OTIB-0014), internal doses of construction workers may be bound based on the tritium urinalysis reporting levels.

7.1.1.2 Uranium

The production of desired materials from the SRS nuclear production reactors required a combination of fissionable material to provide the neutrons (or the "fuel") and fertile material to serve as targets. The basic fuel used in the SRS reactors was U-235. Basic targets consisted of U-238, which would be turned into plutonium. The uranium fuel or targets were manufactured at SRS in the Manufacturing Area (also known as M Area or 300 Area). The bare uranium slugs were typically made at Feed Materials Production Center and shipped to SRS for canning in the Fuel Slug Manufacturing Building (SRS at Fifty, 2002). The building was designed for the preparation of natural-uranium slugs and subsequent canning of the slugs in an aluminum sheath for protection from water corrosion. Based on the suitability testing of the material destined for the reactor (conducted at the M-Area test piles), the fuel and target assemblies advanced from solid slugs to tubular assemblies. SRS used nearly all of the Highly Enriched Uranium (HEU) from the recycled uranium from the INEEL in "driver" fuels, which in turn provided high flux. The first tubular production work at SRS was done in the Alloy Building in mid-1956, but this work was soon transferred to the Manufacturing Building, which was built in 1956 and 1957 specifically for the tube manufacture. The recovery of uranium mixtures and enriched uranium occurred in the 200-F and 200-H Areas. H Area employed the "HM" process to recover enriched uranium. Many metric tons of uranium were used during the reactors' operational periods.

Personnel Monitoring for Uranium

Uranium bioassay has been conducted at SRS since the start of radiological operations. SRS technical documentation indicates that for earlier monitoring periods, the designations "enriched" and "depleted" reported for uranium analysis referred to analysis performed by alpha counting or chemical measurement (fluorometric method), respectively; these designations 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. Activity fractions for depleted, recycled, and highly-enriched uranium are listed in the TBD. Procedures for uranium urinalysis are adequately described in the TBD (ORAUT-TKBS-0003). Alpha spectroscopy has been used since 1990 for specific isotopic analysis. Information regarding the reporting levels and detection limits at SRS is included in the SRS TBD.

A Progress Report issued in March 1961 (DPSP-61-1-2) states that analytical procedures require *radioautographing* on NTA slides to obtain the necessary sensitivity for most alpha emitters but that developments in low-level alpha counting indicates that the lengthy (seven days) radioautographing step may be replaced using silicon detectors and counters.

Although not specifically mentioned in the bioassay records, analysis of U-233 would be conducted via the alpha-counting method with the same separation and measurement methodology used for other isotopes. However, no U-233 analytical results have been reported for SRS claimants.

Feasibility of Bounding Uranium Doses

Construction workers exposed to uranium were monitored by urinalysis; therefore, personal bioassay data are available and reported by the site in their individual monitoring records. These data can be used to bound the individual worker's dose from uranium. For those workers who were exposed but might not have been monitored, sufficient data exists so that a co-worker model can be developed based on monitored workers for both enriched and natural/depleted uranium. Although many of the uranium bioassay logbooks are available, the co-worker model being developed will be based on the claimant data in NOCTS. The coding for these data was completed in late August 2008; the model is currently being developed and documented as of this writing. For the development of this model, there are 3796 fluorometric measurements of uranium from 1953 to 1992, 3133 alpha count measurements from 1955 through 1994, and 813 alpha spectroscopy results from 1994 through 2004. Due to the extensive bioassay monitoring records available to NIOSH, it is feasible to develop a co-worker model; therefore, unmonitored doses can be bounded.

7.1.1.3 Plutonium

Plutonium, mainly Pu-239, was a major production material throughout the life of the SRS reactors. Pu-239 was created from U-238 by neutron bombardment in the reactors. When capturing a neutron, U-238 becomes an unstable U-239 isotope that decays to the more stable Pu-239 isotope by a two-step beta-negative decay process with Np-239 as the intermediate decay product (SRS at Fifty, 2002). The irradiated targets were transported to the 200-F Area and 200-H Area for product separation and packaging. The PUREX separation process was employed in F and H areas to recover plutonium, alternating process schedules for plutonium recovery. Wet chemical separations were accomplished in the canyon portion of the canyon building. Final products were solidified for shipment on the B-

Lines, and uranium was solidified for re-use or storage in the A-Lines (SRS at Fifty, 2002). Other minor isotopes of plutonium, (Pu-238, Pu-240, Pu-241) are by-products from the Pu-239 production processes. The production of Pu-239 began in 1953 and continued until 1989. By the early 1960s, the AEC had 14 production reactors - nine at Hanford and five at SRS. At the peak of production, these reactors produced around seven metric tons of plutonium per year. By 1964, the U.S. alone had over 60 tons of plutonium. Between 1955 and 1964, SRS produced approximately one-third of the nation's plutonium (DOE/DP-0137).

During the 1950s and early 1960s, work was done by SRS to prepare radioisotope thermoelectric generators (RTGs) for the space program. The best heat source for the production of electricity in space was proven to be Pu-238. Production of Pu-238 began at SRS in the late 1950s and continued until 1986. During this period, Pu-238 was created by neutron irradiation of Np-237 targets in an increasingly-sophisticated series of reactor assemblies (SRS at Fifty, 2002). Plutonium recovery occurred in the Canyon Building and the B-Lines at the 200-F and 200-H Areas. Production quantities of Pu-238 were in the metric tons as SRS supplied the majority of the Pu-238 source material for the RTG heat sources for the space programs.

Personnel Monitoring for Plutonium

Urinalysis for plutonium was implemented in 1954. Samples were electroplated and activities were determined by gross alpha track analysis of exposed NTA emulsions. An analytical method to detect plutonium activity was implemented in 1959 that used ion exchange to separate plutonium; the minimum detectable activity was about the same as that of the alpha track method. Results were recorded as either Pu or as Pu238/Pu239. This method was used until 1964 when gross alpha analysis using a solid-state, surface-barrier alpha detector was used. In 1981, alpha spectrometry was implemented along with a new co-precipitation technique (although some with alpha track analysis continued until 1990). With the onset of this co-precipitation methodology, bioassay results for Pu-238 and Pu-239 were reported separately. Separation of plutonium with neptunium, actinides, uranium, and strontium from a single sample began in 2001. Alpha-emitting plutonium and neptunium isotopes were electrodeposited and counted by alpha spectrometry on a single planchet. Information regarding the reporting levels and detection limits at SRS is included in the SRS TBD.

Feasibility of Bounding Plutonium Doses

Construction workers exposed to plutonium were monitored by urinalysis; therefore, personal bioassay data are available and reported by the site in their individual monitoring records. These data can be used to bound the individual worker's dose from plutonium isotopes. For those workers who were exposed, but might not have been monitored, a co-worker model can be developed based on monitored workers for Pu-238 and Pu-239. Although many of the plutonium bioassay logbooks are available, the co-worker model being developed will be based on the claimant data in NOCTS. The coding for these data was completed in late August 2008; the model is being developed and documented as of this writing. For the development of this model, there are 6186 results for total plutonium from 1954 to 1990, and 4076 results from 1981 through 2006 for the specific radionuclides Pu-238 and Pu-239. Due to the extensive bioassay monitoring records available to NIOSH, it is feasible to develop a co-worker model; therefore, unmonitored doses can be bounded.

7.1.1.4 Neptunium (Np-237)

Neptunium-237 was the fertile material used for the production of Pu-238. Neptunium targets were manufactured in the 300 Area and were destined for the production reactors. Neptunium also was created when U-235 was irradiated. Therefore, Np-237 remained within the spent fuel matrices until they were processed in the Canyon Building in the 200-F and 200-H Areas and discharged as High Level Radioactive Wastes to the Waste Tanks. The production periods for Np-237 coincide with the reactors' operational periods. However, quantities of Np-237 remain in the Waste Tanks until the present time. The amount of Np-237 handled at SRS was in the many-kilograms range.

Personnel Monitoring for Neptunium

SRS implemented urinalysis for Np-237 in 1959 (ORAUT-TKBS-0003). This method co-precipitated neptunium with calcium-magnesium ammonium phosphate, separated neptunium with ion exchange, followed by autoradiography on NTA emulsion for 10,000 minutes. While the MDA was not given, it was mostly likely similar to the MDA for plutonium alpha autoradiography method in use at the same time. The results were recorded as S.P. (special product). Information regarding the reporting levels and detection limits at SRS is included in the SRS TBD.

In 1964, SRS implemented a new TIOA extraction and electro-deposition methodology. While the method was more time-efficient, the reported level remained the same as that of the previous method. Samples were analyzed by an off-site, commercial laboratory using extraction chromatography resin from 1993 through part of 1996. In 1996, analysis responsibility returned to SRS where alpha spectrometry was used for neptunium analysis. Since 2001, TEVA and TRU resins have used to separate neptunium from actinides and uranium, and alpha spectrometry has continued. Information regarding the reporting levels and detection limits at SRS, and the method for interpreting and applying values for dose reconstruction, are included in the SRS TBD.

Feasibility of Bounding Neptunium Doses

Construction workers exposed to neptunium were monitored by urinalysis; therefore, personal bioassay data are available and reported by the site in their individual monitoring records. These data can be used to bound the individual worker's dose from neptunium. For those workers who were exposed but might not have been monitored, a co-worker model can be developed based on monitored workers for neptunium. Although the bioassay logbooks are available, the co-worker model being developed will be based on the claimant data in NOCTS. The coding for these data was completed in late August 2008; the model is being developed and documented as of this writing. For the development of this model, there are 304 measurements of neptunium from 1960 to 2004. Due to the bioassay monitoring records available to NIOSH, it is feasible to develop a co-worker model; therefore, unmonitored doses can be bounded.

7.1.1.5 Americium (Am-241 and Am-243)

Am-243 was one of the intermediate isotopes created in the progression of transforming Pu-239 to Cm-244 as well as one of the isotopes studied as a possible heat source. It was also used as target material for the production of transplutonium isotopes. Cm-244 is produced through successive neutron captures in Pu-239. The transmutation would take Pu-239 to Pu-240 to Pu-241, followed by

Pu-242, Pu-243, Am-243, Am-244, and finally Cm-244. A production campaign, Curium I, took the progression from Pu-239 to Pu-242. The targets from this irradiation were reprocessed and made into new targets for the second phase of this campaign, Curium II, which transmuted Pu-242 to the product, Cm-244, through Am-243 and Am-244 (SRS at Fifty, 2002). Because Curium II required a much higher neutron flux than ever been attempted, a demonstration program (called the High Flux Program) was implemented prior to the start of Curium II. In addition to supporting the curium program, the High Flux Program also demonstrated the applicability of high neutron flux for the generation of transplutonium isotopes. The target material for the High Flux Program consisted primarily of Pu-242. In addition, three 1-inch-diameter thimbles contained Am-243, Cu-244, and 150 nuclides of 66 elements for nine universities and laboratories (WSRC-MS-2000-00061). Am-243 was generated during the irradiation processes in the High Flux Program and Curium II and was also used as targets for the High Flux Program. The first production period for Am-243 began in 1959 and continued through 1965. The production picked up in 1967 after a year of stoppage. The early SRS transplutonium programs generated 930 grams of Pu-242, 300 grams of Am-243, and 330 grams of Cm-244. By the end of the program, a total of 5.9 kilograms of curium and many kilograms of Am-243 were generated (SRS at Fifty, 2002). Americium-241 is generated by the decay of Pu-241 and coexists in equilibrium with Pu-241 as its progeny.

Personnel Monitoring for Americium

Records of i*n vitro* bioassay for trivalent actinides (americium, curium, and californium) showed urinalysis data back at least to about 1963. Bioassay for Am-243 was not needed before 1963. DPSP-55-454-002 states: “Recovery of Pu242, Am243, and Cm244 – Fuel for the first transplutonium separation campaign consisted of 13 assemblies discharged from R-reactor in Apr 1963 after approx 5.2 years’ irradiation.” Reporting levels for gross alpha determination of americium varied through 1970. In 1971, gross alpha counting using a solid-state detector was implemented. SRS changed the procedure for radiochemical processing in 1990 Alpha spectrometry has been used since 1994. Up through the implementation of alpha spectroscopy in 1993, exposure location or other information concerning an intake was useful in the determination of the principal radionuclide since the results were obtained for trivalent actinides. Without sufficient information to separate activities to particular radionuclides, the entire activity was assigned to Am-241. Information regarding the reporting levels and detection limits at SRS is included in the SRS TBD.

Feasibility of Bounding Americium Doses

Construction workers exposed to americium were monitored by urinalysis; therefore, personal bioassay data are available and reported by the site in their individual monitoring records. These data can be used to bound the individual worker’s dose from americium.

For those workers who were exposed but might not have been monitored, a co-worker model can be developed based on monitored workers for americium. Although many of the bioassay logbooks are available, the co-worker model being developed will be based on the claimant data in NOCTS. The coding for these data was completed in late August 2008; the model is being developed and documented as of this writing. For the development of this model, there are 926 trivalent measurements from 1963 to 1994, and 96 alpha spectroscopy measurements from 1993 through 2006. Due to the extensive bioassay monitoring records available to NIOSH, it is feasible to develop a co-worker model; therefore, unmonitored doses can be bounded.

7.1.1.6 Curium (Cm-244)

In late 1963, Cm-244 was sought as a possible heat source for the thermoelectric generation of electricity, and it was also desired as a target material for the production of californium. Curium-244 is produced through successive neutron captures in Pu-239. Gram quantities of Cm-244 were first produced in SRS reactors in 1962. In May 1964, the curium production campaigns (Curium I and II) began irradiation to produce kilogram quantities of Cm-244 for evaluation as an alternative to Pu-238 and for use as target material to produce Cf-252. The targets from Curium I irradiation were reprocessed and made into new targets for the second phase of this program (Curium II) to complete the transmutation of Pu-239 to Cm-244. A separate High Flux Program or Demonstration was started in February 1965 and transmuted much of its Pu-242 targets to Cm-244. By 1967, a total of 5.9 kilograms of Cm-244 had been made during the two curium production campaigns. Continued irradiation of plutonium target material eventually produced about 12 kilograms of Cm-244 (WSRC-MS-2000-00061).

Personnel Monitoring for Curium

Records of *in vitro* bioassay for trivalent actinides (americium, curium, and californium) showed urinalysis data back at least to 1963. In 1971, gross alpha counting using a solid-state detector was implemented. SRS changed the procedure for radiochemical processing in 1990. Alpha spectrometry has been used since 1994. Up through the implementation of alpha spectroscopy in 1994, exposure location or other information concerning an intake was useful in the determination of the principal radionuclide since the results were obtained for trivalent actinides. Without sufficient information to separate activities to particular radionuclides, the entire activity can be assigned to Cm-244. Information regarding the reporting levels and detection limits at SRS is included in the SRS TBD.

Feasibility of Bounding Curium Doses

Construction workers exposed to curium were monitored by urinalysis; therefore, personal bioassay data are available and reported by the site in their individual monitoring records. These data can be used to bound the individual workers dose to curium. For those workers who were exposed but might not have been monitored, a co-worker model can be developed based on monitored workers for curium. Although many of the bioassay logbooks are available, the co-worker model being developed will be based on the claimant data in NOCTS. The coding for these data was completed in late August 2008; the model is being developed and documented as of this writing. For the development of this model, there are 926 trivalent measurements from 1963 to 1994, and 92 alpha spectroscopy measurements from 1994 through 2006. Due to the extensive bioassay monitoring records available to NIOSH, it is feasible to develop a co-worker model; therefore, unmonitored doses can be bounded.

7.1.1.7 Californium (Cf-252, including Cm-248 daughter)

Cf-252 is one of the heaviest of the transplutonium elements and produces an intense neutron source from spontaneous fission. The program to specifically create californium (Californium I) began in August 1969 and continued until November 1970, with irradiation work conducted in K Reactor. The program generated the element in sufficient quantities to make the isotope available to industry and medical facilities. The production of Cf-252 entailed the addition of eight neutrons to the nucleus of the starting element, Cm-244, with the atomic number increasing from 96 to 98. The residues provided by Curium II were used as targets for the production of californium. When the program ended in November 1970, SRS had produced 2.1 grams, the largest amount of californium ever produced (SRS at Fifty, 2002).

Personnel Monitoring for Californium

Records of *in vitro* bioassay for trivalent actinides (americium, curium, and californium) showed urinalysis data back at least to about 1963. In 1971, gross alpha counting using a solid-state detector was implemented. SRS changed the procedure for radiochemical processing in 1990. Alpha spectrometry has been used since 1994. Up through the implementation of alpha spectroscopy in 1994, exposure location or other information concerning an intake was useful in the determination of the principal radionuclide since the results were obtained for trivalent actinides. Without sufficient information to separate activities to particular radionuclides, the entire activity can be assigned to Cf-252. Information regarding the reporting levels and detection limits at SRS is included in the SRS TBD.

Feasibility of Bounding Californium Doses

Construction workers exposed to californium were monitored by urinalysis; therefore, personal bioassay data are available and reported by the site in their individual monitoring records. These data can be used to bound the individual workers dose to californium. For those workers who were exposed but might not have been monitored, a co-worker model can be developed based on monitored workers for californium. Although all of the bioassay logbooks are available, the co-worker model being developed will be based on the claimant data in NOCTS. The coding for these data was completed in late August 2008; the model is being developed and documented as of this writing. For the development of this model, there are 926 trivalent measurements from 1963 to 1994, and 92 spectroscopy measurements from 1994 through 2004. With this bioassay monitoring records available to NIOSH, it is feasible to develop a co-worker model; therefore, unmonitored doses can be bounded.

7.1.1.8 Thorium (Th-228, Th-232, and U-233)

The details of the thorium processes and programmatic progress can be gleaned from the Monthly Progress Reports of the Works Technical Department beginning with the January 1953 edition and unfolding over subsequent years. The Feed Materials Production Center (FMPC) was the principal supplier of thorium to SRS. Much of that was canned by Sylvania Electric Products (SEP) Company with subsequent fuel fabrication with a small fraction at SRS.

In August 1954, Test Authorization Number 3-138, *DPSOX-00530, AlSi Canning Thorium Slugs* (Huntoon, 1954), in Building 313-M was approved by the Plant Manager, SRL Director and seven others, including the Health Physics Superintendant. By November 1954, four quatrefoil fuel assemblies had been produced and were included in the L-2 reactor loading for irradiation testing (DPSP 54-1-10; DPSP 54-1-11). The plan was to return the slugs to the hot labs of R or P areas when this test was completed (DPSP 54-1-11); in May 1955, four irradiated thorium slugs were returned to SRL for testing in the 100-R hot lab (DPSP 55-1-8-DV).

Early in 1955, Sylvania Electric Products (SEP) was contracted to produce 11,000 hot-press bonded thorium slugs due to their demonstrated capabilities to produce uranium slugs using the same process and the low yields from the competing AlSi hot-dip process at SRS. A second order for 15,000 more canned thorium slugs was largely fulfilled in August 1955 (DPSP-55-454-3). The SEP thorium fuel was used for the R-7 full-core reactor loading (DPSP 55-1-8-DV). Programmatic changes ended the need for thorium fuel. In September, the SEP thorium program was reviewed, resulting in a plan to dispose of the unused inventory by stripping the cans and returning the metal to FMPC (DPSP 54-1-11)

In September 1956, SRS received a request to irradiate five tons (approximately 5000 slugs) of thorium to produce U-233 (DPSP 56-1-9). Shipping continued in August and a plan was presented to retain two irradiation thorium slugs from R-11 to make activity measurements at the test facility in 105-K Disassembly Area during the following year to define the absolute decay schemes associated with the irradiated fuel (DPSP 57-1-8). This research began according to plan in November 1956 (DPSP 57-1-11-DV). In December 1956, Building 313-M personnel were reduced to 109 wage roll and 21 supervisors from the previous level of 154 wage roll and 26 supervisors (DPSP-55-454-3).

In December 1958, ORNL proposed that pellets of mixed 4% enriched UO_2 and ThO_2 be irradiated at SRS. It is not known if this occurred; however, in June 1959 a procedure was developed and tested for making a final arc weld on aluminum-canned hot-press bonded slugs containing powdered and compacted thorium oxide.

In October 1961, irradiated metal waste, including thorium and uranium with 8400 Ci of fission products was shipped from the SRL high-level caves to Building 643-G. In 1962, Bridgeport Brass Company extruded 43,000 kg of thorium metal which was machined and canned at Sylcor. In February 1963, 6000 Mark VII-T canned thorium slugs were finished at SRS for use in the L-1 charge. Then 10,000 more were finished in November 1963 for use in the L-2 charge.

In 1964, vibratory compaction equipment was installed to produce thorium oxide (thoria) slugs. By August 1964, 13,600 slugs (each containing 3.6 kg of thoria) were being produced at a rate of 120 per shift. It was noted that contamination was controlled satisfactorily (DPSP-55-454-3). Also in 1964, Sylcor canned and delivered 9600 Mark VII-TS solid thorium slugs. In 1965, there were 28,160 thorium oxide slugs canned (DPSP-55-454-3). In 1966, thoria slug fabrication facilities in Building 313-M were improved in preparation for a new production campaign. In 1967, process development continued until March when operator training began (DPSP-55-454-3). During 1967, there were 23,962 Mark 50A slugs produced (each containing 3.7 kg of thoria), and 17,098 Mark 50B slugs (each containing 1.7 kg of thoria). All but 2,100 of these were produced between April and August in three-shift operations. During November and December, the remaining 2100 slugs were produced on a single-shift schedule (DPSP-55-454-3). In January and February 1968, an additional 3,760 thoria slugs were produced, completing the campaign (DPSP-55-454-3)

Personnel Monitoring for Thorium

Urine bioassays for thorium were conducted in accordance with a procedure (Ellett, 1958), and work technical report (DPSP 56-1-8). A re-sampling threshold was established at SRS as the bioassay program matured (Logbook P006, p. 78). A 1956 bioassay logbook dedicated to thorium bioassays shows that 224 bioassay were performed on 168 different individuals. The bioassay program was carefully done, as evidenced by the 42 blanks and 45 spikes recorded in the logbook. With minor exceptions, the net sample results were similar to the sample blanks. At least one additional thorium bioassay result predating the logbook has been found in a claimant's dosimetry record, suggesting that more samples will be discovered when earlier logbooks are examined. No urine bioassay samples for thorium have been found after 1956. This date roughly coincides with the end of bare thorium metal canning at SRS, which may provide an explanation. Contemporary ICRP biokinetics models for thorium show that urine bioassay lacks the sensitivity to demonstrate compliance with regulatory limits, and therefore, urine bioassay is not used as part of a routine thorium bioassay program. The lack of information on thorium biokinetics was mentioned in the procedure (Ellett, 1958). The Internal Dosimetry Technical Basis Manual (WSRC-IM-90-139, Rev. 1) identifies bioassay methods for Th-228 and Th-232 which would have been performed after 1960.

In-vivo counting was implemented in 1960 and would have been sensitive to thorium daughter products. Whole body counting and chest counting were routinely used as part of the internal dosimetry program at SRS.

Airborne Monitoring for Thorium

Representative air sampling data have been requested for locations where un-encapsulated thorium was processed prior to 1960. ICRP 75, *General Principles for the Radiation Protection of Workers*, discusses the uses of area air sampling to establish trends and representative air sampling to quantify the extent of airborne contamination to which workers are likely to be exposed. The air sampling requirement specified in the August 1954 Test Authorization Number 3-138 suggests that the air sampling data will satisfy the "representative sampling" definition in ICRP 75 which states that representative sampling typically involves using fixed samplers at locations intended to be reasonably representative of the breathing zone of the workers. If air sample data are available as expected, then intake rates can be calculated based on an assumption of a default breathing rate and an appropriate occupancy factor in these areas using the air sample data.

Feasibility of Bounding Thorium Doses

Some construction workers exposed to thorium were monitored by urinalysis in 1956 and other workers were monitored by *in vivo* analysis after 1960. Using the *in vivo* data, doses received from potential exposures to thorium starting in 1960 can be bounded for production workers, and therefore, for construction and maintenance trade workers by a combination of:

- *in vivo* counting data that became available after 1960; and
- representative air sampling data from locations where un-encapsulated thorium was processed prior to 1961.

NIOSH is reviewing the 1956 bioassay data to determine the adequacy of that data for assigning internal dose. At this point, NIOSH has not determined whether the reconstruction of thorium doses received prior to 1960 is feasible. A follow-up to this report will be issued once the air sample data prior to 1960 has been reviewed.

7.1.1.9 Fission and Activation Products

Europium-152/154 was produced primarily as fission products in the fuel and also by neutron activation of reactor control rods. Europium, along with other fission products, was separated in the 200-F and 200-H Areas from the product streams containing uranium and plutonium. Because Eu-152 had useful application as a food sterilizer and as a source for radiation therapy, SRS isolated Eu-152 from the Purex waste stream for evaluation in a 1967 campaign. It has been estimated that kilograms of Eu-152 were produced at SRS.

In SRS reactors, a small percentage of the uranium in the fuel was converted into other fission products as by-products of energy generation. Activation products resulted from neutron capture by other materials, such as structural components of the reactor, the reactor coolant, control rods, or fuel and target claddings and containers. The dose-significant fission and activation products at SRS include Mn-54, Ce-144, Zn-65, Sr-90, Ru-106, Sb-125, and Cs-134/137. Fission and activation products remain in the fuels and control rods until they are reprocessed and separated from uranium and plutonium at the Canyon Building in the 200-F and 200-H Areas and discharged to the Waste Tanks as high level radioactive wastes. Production of these radionuclides coincides with the reactors' operational periods. However, metric tons of these radionuclides currently remain in the Waste Tanks.

Personnel Monitoring for Fission and Activation Products

Initially, intakes of gamma-emitting fission products were monitored by urinalysis, which began in 1954 when the reactors went on line. The start date for urinalysis for strontium radioisotopes was not determined more specifically than the "late 1950s." However, Boni discusses a bioassay procedure in use at the time for rapid estimation of the total amount activity of beta-gamma-emitting isotopes in a urine specimen with recoveries of Sr-89, Sr-90/Y-90, Zr-95/Ni-95, Ce-144/Pr-144, Fe-59, Cr-51, and Zn-65 greater than 90% and Co-60 of 85% (Boni, 1959). Fission product bioassay was performed to determine intakes of Tm-170.

Welders and grinders at completion of their work in the (reactor RDZ) area, but prior to leaving the area, were required to submit urine samples for induced activities (750 mL), in addition to tritium samples as of August 26, 1957. The sample was to be started soon enough for completion prior to leaving the area (e.g., approximately ten days before) (Health Physics, 1957).

Gamma-emitting fission products were monitored by urinalysis prior to whole body counts, but also into the mid-1960s. Strontium isotopes (and potentially cerium and promethium) were separated by alkaline earth phosphate co-precipitation followed by counting on a GM counter. This procedure was in use from start-up through 1969 when replaced by liquid ion exchange methods that separated the yttrium progeny and beta proportional counting. Since 2001, strontium was separated as part of the use of TRU and TEVA resins for plutonium, uranium, actinides, and strontium (ORAUT-TKBS-0003). The procedure separated and counted radionuclides of Sr, Y, Ce, and Pm. Analyses for beta and for gamma-emitting fission and activation products have been referred to as FPIA (Fission Product Induced Activity).

Whole-body counting has been performed since 1960 when the SRS Whole Body Counting Facility was completed. A March 16, 1961 report (DPSP-61-1-2) stated that the body burdens for 84% of 37 Separations employees in the 200-F Area, measured with the whole body counter, had detectable amounts of Ce/Pr-144, Ru/Rh-106, or Zr/Nb-95. The maximum amounts found were 21 nanocuries Ce/Pr-144, 32 nanocuries Ru/Rh-106, and 1.8 nanocuries Zr/Nb-95. Minimum detectable activities for *in vivo* analysis of fission and activation products are given in ORAUT-TKBS-0003 (pp. 72-73).

Airborne Monitoring for Fission and Activation Products

SRS performed air sampling through 1956 to determine if respiratory protection should be used. The requirement to wear a respirator was based on the measured level of Fe-59. Due to multiple slug failures during 1957, experimental and developmental programs in disassembly areas, and interchange of disassembly equipment between areas, the masking requirements basis was changed to unknown fission products.

Feasibility of Bounding Fission and Activation Product Doses

Construction workers exposed to fission and activation products were monitored by either urinalysis or whole-body counting or both; therefore, personal bioassay data are available and reported by the site in their individual monitoring records. These data can be used to bound the individual worker's dose from fission and activation products. For those workers who were exposed but might not have been monitored, a co-worker model can be developed for fission and activation products based on monitored workers. Although bioassay logbooks are available, the co-worker model being developed will be based on the claimant data in NOCTS. The coding for these data was completed in late August 2008; the model, which will provide intakes of the radionuclide providing the highest organ dose, is being developed and documented as of this writing. With more than 4,700 results in NOCTS, it is feasible to develop a co-worker model; therefore, unmonitored doses received from exposures to fission and activation products can be bounded.

7.1.1.10 Cobalt-60 (Co-60)

Co-60 was produced by the irradiation of natural cobalt (Co-59) in the SRS reactors. Cobalt-60 emits very penetrating gamma radiation. Sources made of Co-60 can be used for cancer treatment, food sterilization, radiography, and as heat sources for thermoelectric generators. Small amounts of low-specific-activity Co-60 (~50 curies per gram) were produced in SRS reactors as early as 1955. Beginning in 1965, the High Flux Program (see the Cm-244 description), provided an opportunity to make large quantities of very-high-specific activity Co-60 when Co-59 was the preferred material for control and safety rods in high flux reactor cores (WSRC-MS-2000-00061). The first production program (1955-1958) made Co-60 with a specific activity of 50 Ci/g. The second period of production (1959-1964) created Co-60 with 100 Ci/g. The third program (1964-1967) generated material with 700 Ci/g. The cobalt produced at SRS is estimated to be in the kilogram range (SRS at Fifty, 2002).

Personnel Monitoring for Cobalt

Exposures to Co-60 have been monitored using *in vivo* analysis (whole body counting) since about 1960. Prior to that, and continuing through the mid- to late-1980s, gamma-emitting fission products were monitored by urinalysis using both gross beta analysis and gamma-spectrometry analysis. NIOSH has obtained the logbooks for many urinalysis results. See the discussion on fission/activation product bioassay under Section 7.1.1.9.

Feasibility of Bounding Cobalt Doses

Construction workers exposed to cobalt were monitored by urinalysis and/or whole body counting; therefore, personal bioassay data are available and reported by the site in their individual monitoring records. These data can be used to bound the individual worker's dose from Co-60. For those workers who were exposed but might not have been monitored, a co-worker model can be developed based on monitored workers for cobalt. Although all of the bioassay logbooks are available, the co-worker model being developed will be based on the claimant data in NOCTS. The coding for these data was completed in late August 2008; the model is being developed and documented as of this writing. For the development of this model, more than 4700 measurements from 1954 to 2004 for fission product determination are available. Due to the extensive bioassay monitoring records available to NIOSH, it is feasible to develop a co-worker model; therefore, unmonitored doses can be bounded.

7.1.1.11 Polonium

Starting in June 1966, Po-210 was produced at SRS by the irradiation of bismuth slugs as part of the Curium II project (DPSP-67-1281). SRS produced 324,200 curies from June 1966 through 1967. The slugs were shipped to Mound in Ohio. Recovery of polonium took place at Mound, not at SRS.

From 1967 through 1970, SRS performed research using laboratory and plant-scale equipment to test the engineering feasibility of using liquid metal distillation to separate up to 10 kg of Po-210/year from irradiated bismuth (DP-1222). Long-lived (>60 day half-life) gamma-emitting impurities had to be separated from the Po-210 distillation. While the percentage-by-weight of those impurities was small, SRS felt that they could result in unacceptable levels of gamma radiation. Some separation was achieved by alloying for decay. A high-purity grade of bismuth was irradiated and then cooled for about 15 months. After the cool-down, Zn-65 was distilled with Po-210, while some Sb-124 and Co-60 activity remained in the bismuth residues.

Personnel Monitoring for Polonium

As noted above, the research conducted in 773-A was a very small scale. NIOSH has obtained bioassay logbooks for analysis of Po-210 and believes the individuals involved in the project were monitored via bioassay. NIOSH has requested the bioassay procedure used to interpret the results. Together, the data and procedure will be used to bound internal doses received from exposure to polonium for these few workers.

Feasibility of Bounding Polonium Doses

NIOSH can bound potential internal exposures received from polonium using the available bioassay data. NIOSH will use this limited data set to develop a co-worker intake model for use in bounding doses. For construction workers who may not have been monitored, internal exposures received from potential intakes of polonium can be bounded through the application of the co-worker model.

7.1.2 Internal Dose Reconstruction Feasibility Conclusion

The guidance of ORAUT-OTIB-0052 is to determine internal dose of construction workers using the same method applied to all other SRS workers.

The dose to SRS construction workers who were monitored for internally-deposited occupationally-related radionuclides may be bounded with sufficient accuracy based on their reported bioassay monitoring results.

Additionally, workers who were not monitored, or whose records provided by the Department of Energy may be incomplete (e.g., missing data), may be assigned internal doses based on information in Section 4.4.3 of the SRS TBD (ORAUT-TKBS-0003) and from intakes derived in co-worker models discussed in Section 7.1 of this Evaluation Report.

Doses received from potential exposures to thorium starting in 1960 can be bounded using the data and methods discussed in Section 7.2.1.5. The feasibility of bounding radiation doses received from exposure to thorium from start-up through 1959 is being investigated. That determination is reserved.

7.2 External Radiological Exposure

ATTRIBUTION: Section 7.2 and its related subsections were completed by Mike Mahathy, Oak Ridge Associated Universities; Eugene Potter, Mel Chew and Associates, Inc; Bryce Rich, Mel Chew and Associates, Inc.; and Darin Hekkala, Dade-Moeller, Inc. These conclusions were peer-reviewed by the individuals listed on the cover page. The rationales for all conclusions in this document are explained in the associated text.

SRS was built to produce the materials used in the fabrication of nuclear weapons, primarily tritium and Pu-239. Five reactors were built to produce nuclear materials through irradiation of target materials with neutrons. Support facilities included two chemical separations plants, a heavy water extraction plant, a nuclear fuel and target fabrication facility, and waste management facilities. HEU fuels were used within the reactor core to create higher neutron flux areas, which in turn allowed more efficient production of other special isotopes (i.e. Pu-238, Cf-252, and others). In addition, the Savannah River Laboratory conducted research in support of the production operations and special isotope production campaigns. In the following subsections, the source term, personnel monitoring methods, and feasibility of reconstructing photon and neutron doses are discussed.

7.2.1 Photon Exposures

7.2.1.1 Source Term

The photons associated with each of the three main steps of material production are discussed below along with exposures experienced at the Savannah River Laboratory. The three main steps in nuclear material production include the fuel and target fabrication, reactor operations, and chemical separations.

300 Area - Fuel and Target Fabrication

The primary target materials loaded into the SRS reactors consisted of depleted uranium for Pu-239 production. These targets were manufactured in the fuel and target fabrication area (300 Area). The photon energy spectra from uranium covers both intermediate and high-energy photons; both are easily measured using film badges and TLDs. The external photon dose rate from this source was relatively low so work around the targets before irradiation did not require any special shielding and workers could generally handle the fuel and targets wearing gloves. During special isotope production campaigns, the photon energy spectra and dose rates could be significantly different compared to the standard depleted uranium work, with increased low-energy photons for some target materials to which the film badge responded with significant increase in indicated dose response. For these campaigns, it is evident from the dosimetry records and the monthly works technical reports that the SRS Health Physics Group was aware of these differences and implemented a special method for interpretation of the darkening on the badge film by using a low-photon energy calibration curve to account for the difference in photon energy spectra.

100 Area – Reactor Operations

At the reactor areas, the fuels and targets were loaded into assemblies and then placed in the reactor. The reactor was then operated for a set time period depending on the isotopic production goal of the irradiation. During operation, biological shields around reactor core reduced the dose levels from prompt fission photons and fission product photons from irradiated fuel. The top of the reactors were at grade level and access to the lower levels was restricted/controlled during operation. After the fuel and targets were irradiated in the reactors, they were removed and allowed to decay in the spent fuel pool or disassembly basin area of the reactor. The transfer of the fuel and targets from the reactor to the disassembly area was conducted remotely because the mixed fission products and activation products made these targets highly radioactive so that shielding was necessary to protect workers, minimize exposures, and maintain levels below regulatory limits at maximum. Once in the disassembly area, the spent fuel and targets were maintained in a pool of water that had the dual function of cooling the thermally-hot targets and shielding workers from photon radiation. While the photon emissions associated with these irradiated fuels and targets covered the whole energy spectrum, the predominate photon energies were greater than 250 keV. This energy range is considered "high energy" and is relatively easy to measure using film dosimeters and TLDs. With very few exceptions, all workers exposed to these photons were monitored for external exposure.

200 Area – Chemical Separations (Canyons) Operations

The irradiated fuels and targets were sent to the chemical separations areas via shielded containers on railcars for nuclear material (product) extraction. Once in the canyons, the irradiated targets were dissolved and materials of interest were separated. The uranium was recovered to be "recycled" in the A-line; the plutonium was recovered in the B-Line; and the remaining mixed fission and activation products went to the waste tanks (Tank Farms). As discussed above, uranium is relatively evenly split between intermediate and high-energy photons and the highly-radioactive waste predominately consisted of high-energy photons. Photons associated with recovered plutonium are generally of low energy; however, when quantities of Pu are concentrated in a workplace, moderate dose rates can result. The photon dose rate from plutonium is not nearly as high as the waste material, but is significantly higher than uranium due to higher specific activity and increased neutron emission rates.

700 Area – Savannah River Laboratory

The photon exposures and associated energy spectra at the Savannah River Laboratory was complex and changed over time due to multiple operations, special isotope campaigns, and other research that was conducted. However, the film dosimeters and TLDs were capable of measuring the full photon energy spectra (low, intermediate, and high).

For more information on photon sources and the energy spectra associated with SRS operations, see ORAUT-TKBS-0003, Table 5.3.4.1-1.

7.2.1.2 Personnel Monitoring Data

SRS had an extensive radiation safety monitoring program to measure and control external photon radiation exposure in the workplace. Measurements were conducted using portable radiation instruments, pocket ionization chambers, and personnel dosimeters. Control of external radiation hazards was established through the implementation of Special Work Permits (SWPs) and the establishment of radiation control areas. The predominant method of personnel radiation exposure monitoring was the individual dosimeter. The first beta/gamma dosimeters used at the onset of operations in November 1951 were provided by the Oak Ridge National Laboratory (ORNL). ORNL provided beta/photon film dosimeters and neutron nuclear track, type A (NTA) emulsion dosimeters until SRS implemented an in-house dosimetry program in March 1953 (WSRC-RP-95-234). SRS dosimetry methods evolved over the years as improved technology was developed and the complex radiation fields were better understood. The initial personnel dosimeter program consisted of a two-element film dosimeter capable of measuring low, intermediate and high energy photons; however, the dosimeter over-responded to low and intermediate energy photons (Wright, 1958).

The early (pre-1958) photon dose measurements for SRS employees were hand-entered onto cards and filed in individual employee folders. Each SRS employee has a folder that contains this early monitoring data. For some construction trades workers, however, the monitoring data were recorded on 3x5 visitor cards and cycle monitoring log sheets that are filed by time and by area. During this evaluation, these visitor cards and cycle monitoring data were located and requested from the site. They are undergoing classification review as of the writing of this report. When a claim is filed by an SRS worker, in addition to searching their regular records, the SRS Radiation Records group also searches the visitor cards for the individual worker. NIOSH is currently verifying that the information on the monitoring sheets matches the visitor cards and that these are received for each monitored claimant. This verification is expected to be completed in February 2009. At this time, NIOSH does not have any reason to believe that these monitoring records are different from the information included in each individual claim record.

In November 1959, a new multi-element film dosimeter and automatic film processor was introduced (WSRC-RP-95-234, Davis 1960 DP-471). The use of the multi-element filters improved the low and intermediate energy response (i.e., they better corrected for the lower-energy over-response). The data from this dosimeter as well as the data from the last year from the previous dosimeter (1958) were entered into a new IBM computer system for dose record tracking. On a quarterly basis, SRS produced a quarterly printout of all monitored workers. In addition to the standard printout, a special printout of the people who exceeded the administrative threshold for the quarter, and for those who were projected to exceed the annual limit of 5 rem per year (i.e., for the second quarter the annual total was increased to 2.5 rem, and for the third quarter the annual total was increased to 3.75 rem.) During this evaluation, NOSH has obtained a copy of all of the quarterly dosimetry printouts between 1958 and 1988.

In April 1970, TLDs replaced film badge dosimeters at SRS. This style dosimeter has been, and still is, the standard for external dose monitoring. The record-keeping system (IBM Quarterly Printouts) remained the same. In 1979, the Health Protection Annual Radiation Exposure History (HPAREH) system was initiated. At that time, all of the current employees' annual radiation exposure records were hand-entered into this new electronic system. The subsequent annual doses were then transferred each year to HPAREH. This continued until the initiation of a fully-electronic system in April 1989 called the Health Protection Radiation Exposure Database (HPRED).

As indicated above, individual radiation dose records from personnel dosimeters worn by workers and co-workers are available from 1951 to present. Figure 6.1 illustrated that on the order of 6000 workers per year were routinely monitored for radiation exposure. The SRS Health Physics department required dosimeters be issued to all workers who entered radiation controlled areas. During NIOSH review of SRS documentation, an exception to this monitoring practice was noted in a Test Authorization (DPSOX-254) from November 1953 to July 1955. In this authorization, only persons assigned to a particular location were assigned film badges (i.e., visitors were not always assigned badges). The restriction on this authorization is that only work with natural uranium was allowed and the practice was only allowed until dosimeters were available at the area gate house. This Test Authorization was a stopgap between the start-up period and the regular issuance of film badges from the area gate houses at the access point to the specific area. The Test Authorization was rescinded in July 1955 because it was no longer needed.

Another type of monitoring exception noted in the SRS Radiation Survey Log Sheets (RSLS) is that for some construction work, access controls were put in place to limit the exposure received by construction workers during large modifications. The access controls effectively cordoned off the work zone within a larger radiation control area. Within this area, radioactive materials were removed from the work zone so that the only external exposure would originate from outside the work zone. The perimeter of the work zone was then monitored through the use of survey instruments and/or dosimeters hung on the perimeter towards the radioactive materials to ensure that radiation levels within the zone were below regulatory requirements. As noted, these exceptions are documented in the Radiation Survey Log Sheets (RSLS).

Thus, while the normal practice was to monitor all individuals within a radiation control area, there were occasions, as indicated by workers during the outreach meetings, in which it would appear as if the workers were not monitored within a radiation area. However, based on NIOSH's review, area monitoring was being conducted to ensure that exposures were less than regulatory requirements. Since this control was in effect, the use of a co-worker model for workers within these controlled zones supports NIOSH's ability to bound the photon dose for the class under evaluation.

7.2.1.3 Feasibility of Estimating or Bounding Photon Dose

The SEC evaluation effort was focused on the ability to bound dose for the class (unmonitored construction trade workers). At SRS, construction trades workers (CTW) were generally monitored for radiation exposure just like other production workers; when they entered a radiological controlled area, they were issued a film badge dosimeter (or TLD in later years). However, there are a few occasions where a construction worker might not have been monitored and radioactive materials could have been present (e.g., 1953-1955). However, as discussed in the preceding sections, these doses can be bounded using co-worker monitoring data.

Dose reconstructions are performed for unmonitored CTW at SRS using co-worker dose data provided in ORAUT-OTIB-0032. The application of co-worker data for all dose reconstructions (for unmonitored periods) is described in ORAUT-OTIB-0020. Guidance in ORAUT-OTIB-0052 is applied in concert with ORAUT-0TIB-0020 for unmonitored CTW. ORAUT-OTIB-0052 recommends that a correction factor of 1.4 be applied to co-worker doses used when reconstructing doses for unmonitored CTW. In other words, the dose is increased 40% over the dose that would be applied to an unmonitored non-CTW determined to have been routinely exposed to radiation.

As noted previously in Section 6.2, the co-worker annual dose data provided in ORAUT-OTIB-0032 uses the HPAREH database, which is known to be incomplete for years prior to 1979. However, NIOSH has reviewed the HPAREH database and compared it to a sample of the full hardcopy data from the Quarterly Dosimeter Reports and found that the dose distributions in HPAREH may be used to bound the general population's annual photon dose and, therefore, can be used to bound individual worker photon exposures.

7.2.2 Neutron Exposures

Neutron exposures at SRS were limited to specific operations within several areas. The sources of neutron exposure are shown in Table 7-3 and are limited to areas such as reactor operation and any process with significant quantities of transuranics (TRU) (i.e., with spontaneous fission and α/n reaction potential) or other high neutron-emitting sources. The 200 and 700 areas were the most important with regard to neutron exposures. While one might generally consider neutron exposures to be significant around the production reactors, the exposure potential was actually quite low due to extensive shielding. Another area with intermittent neutron exposure was the fuel and target fabrication area (300 area). The exposures in this area depended on the campaign being conducted. In the subsections below, the sources of neutrons, the personnel and area monitoring that was conducted, and the feasibility of reconstructing neutron doses are presented.

Table 7-3: Sources of Neutrons

Source	Dates	Exposure Potential Comments
Five Production Reactors	R 12/53 – 6/64 P 2/54 – 8/88 K 10/54 – 7/92 L 7/54 – 6/88 C 3/55 – 6/85	Operating reactors extensively surveyed during start-up – essentially no neutron exposure potential. Monitoring for neutrons was performed in preparation for occasional entries during operation to equipment inside lesser-shielded cubicles. Neutron personnel dosimetry was provided. Typically 1:10 n/p ratios were measured.
F Separation Plant plus A, B, and JB product lines; H Separation Plant plus HB product lines and NSR Facility	F 11/54 – 1/57 2/59 – 1989 H 7/55 – 3/59 5/59 – 1989	TRU isotopes were separated and processed. Neutrons were produced through spontaneous fission and α/n reactions. Moderated spectra of varying degrees were encountered. Facilities were shut down for remodeling and process change during the early years. Surveys and personnel neutron dosimetry were provided, as needed.
R&D, Special Product, Technology, Vault & Storage Facilities, uranium target and fuels fabrication	PuFF and PEF 1978-1989 773-A, 313M, 320M, 321M, 322M, 341M 1952 - 1989	Quantities of TRU processed and/or stored 773-A: Cf-252 product 300 Areas: Primarily uranium target and fuels fabrication and development
Analytical and Calibration Facilities	A-772-F and 772-1F production control labs. 736-A Calibration Facility 1953 - 2003	Smaller quantities of TRU and variety of neutron sources in the Calibration Facility, PuBe, PuF_4 and Cf-252

7.2.2.1 Source Term

300 Area - Fuel and Target Fabrication

For certain campaigns, such as the production of trans-plutonium isotopes (californium and curium), plutonium-aluminum (Pu-Al) targets were manufactured in the 300 area. In addition, for the production of Pu-238, neptunium-aluminum (Np-237-Al) alloy targets were manufactured. The neutron dose rate from these targets was low to moderate, with the plutonium being much higher than neptunium from a neutron dose rate standpoint.

During this evaluation, NIOSH found that routine neutron measurements of these targets were conducted and recorded on the Radiation Survey Log Sheets (RSLS) and occasionally reported in the Health Physics Monthly Reports. Table 7-4 presents a few examples of the neutron dose rate measurements on these targets recorded in the 300/700 Area monthly reports.

Table 7-4: Typical Neutron Dose Rates on Special Targets					
Target	**Date**	**Distance and # tubes**	**Fast Neutron Dose Rate (mrem/hr)**	**Total Dose Rate (mrem/hr)**	**N_f to Photon Ratio**
Pu^{239}-Al	January 1971	18" 1-Tube	4	200	0.02
Pu^{239}-Al	January 1971	18" 6-tubes	15	800	0.02
Pu^{242}-Al	June 1969	18" 1-Tube	1	2	1.0
Pu^{242}-Al	June 1969	18" 6-tubes	1	6	0.2
Np^{237}-Al	January 1971	18" 1-Tube	1	80	0.01
Np^{237}-Al	January 1971	18" 6-tubes	2	300	0.01

For the Pu-239-Al and Np-237-Al targets, the vast majority of the external dose came from the photons not neutrons. However, the Pu-242-Al targets illustrated a significantly higher neutron-to-photon ratio.

100 Area – Reactor Operations

As a general rule, neutron doses around the production reactors occurred only while the reactor was operating, with the exception of neutron source handling. As noted in the photon section, the top of the reactors were level with the ground floor; thus, most of the neutron exposure was below-grade. The areas below grade were controlled/restricted during reactor operations. The reactors' biological shielding prevented most neutrons from reaching the general operations areas occupied by workers. From NIOSH's review of the Radiation Survey Log Sheets, there appears to be a couple of areas that personnel were occasionally allowed to enter during reactor operations that could have resulted in neutron exposure. These were the crane wash area and the labyrinth to the pump room. However, as noted in the Radiation Survey Log Sheets, neutron survey measurements were recorded prior to entry and neutron dosimeters were required. Based on a limited review of the Radiation Survey Log Sheets, the general N:P ratio appears to be about 1:10, or 1 mrem of neutrons per 10 mrem of photons.

200 Area - Separations

At the 200 Area plutonium processing facilities, neutron radiation is generated from plutonium or other transuranics by spontaneous fission and/or by alpha particle interaction with light elements such as oxygen, fluorine, and carbon in the product processing lines, which have lesser shielding than the separations canyons. The predominate neutron exposures occurred along the B-lines or plutonium process lines. The three B-lines were the FB Line (early operations), JB Line (later operations), and the HB line. Neutrons were produced through spontaneous fission and α/n reactions. Moderated spectra of varying degrees were encountered at locations along these lines in which there were quantities of TRU/TRP. Throughout the operations, facilities were occasionally shut down for remodeling and process change; thus, neutron monitoring can appear to be intermittent. However, from the review conducted for this evaluation, the monitoring appears to correspond well with production operations.

700 Area – Savannah River Laboratory

The predominate neutron exposures at the Savannah River Laboratory were associated with research being conducted on various plutonium and trans-plutonium isotopes and special materials processing/production. The production of Cf-252 is one of the most notable neutron exposures within this area as californium is a strong neutron emitter due to its relatively high probability (cross-section) for spontaneous fission.

7.2.2.2 Neutron Personnel Monitoring Data

Personnel neutron dose monitoring was provided through the use of dosimeters for workers in areas in which survey data indicated a neutron dose rate greater than 1 mrem/hr, or where the potential for neutron dose rates existed. Neutron monitoring using NTA film began in August 1953 and was replaced by TLND in June 1971, which in turn was replaced by the automated Panasonic UD-809 albedo neutron dosimeter in January 1995. CR-39 track-etch detector type dosimeters were experimentally investigated and researched, but never used to assign personnel exposures.

The number of workers monitored for neutron dose varied with time due to changing radiological conditions, changing standards and regulations, and material production. Because of the restricted areas of the plants in which detectable neutron radiation was present, relatively few workers were monitored for neutron exposure; fewer still members of the CTW class currently under evaluation, since members of this class were generally not routinely assigned work in operating areas in the reactors, TRU production areas, or experimental/R&D areas where significant quantities of neutron-generating materials or neutron sources were present during operational periods.

Figure 7-2 depicts the number of beta-gamma badges and Figure 7-3 depicts the number of NTA badges processed per cycle from the Personnel Meters Reports for the 200-F Area. This area was chosen as an example because this area was one of the highest neutron-exposure areas at SRS. At the time of this writing, the data in these graphs are incomplete due to delays in receiving the Personnel Meters Reports from SRS (i.e., data from 1960 and part of 1962 are missing). The interesting point of these figures is that while the number of photon dosimeter badges remained relatively constant from 1958 through 1970, the number of neutron dosimeter badges increased from 1961, in which less than 2% of the 200-F area workers were monitored for neutron exposure, to near 15% in 1969. In 1970, SRS began to monitor some workers on a quarterly basis while others (more highly-exposed) remained on a monthly basis. Considering this component, the relative fraction of the most highly-exposed workers monitored for neutrons increases from 15% to near 25%.

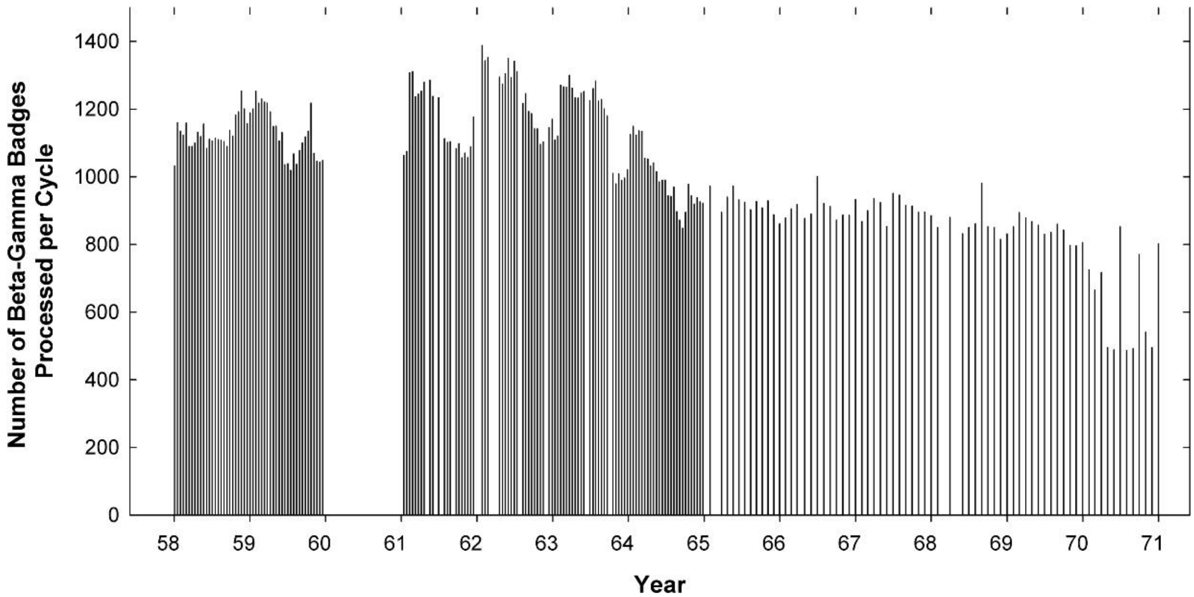


Figure 7-2: Number of Beta/Gamma (Photon) Film Dosimeter Badges Processed per Cycle for the 200-F Area.

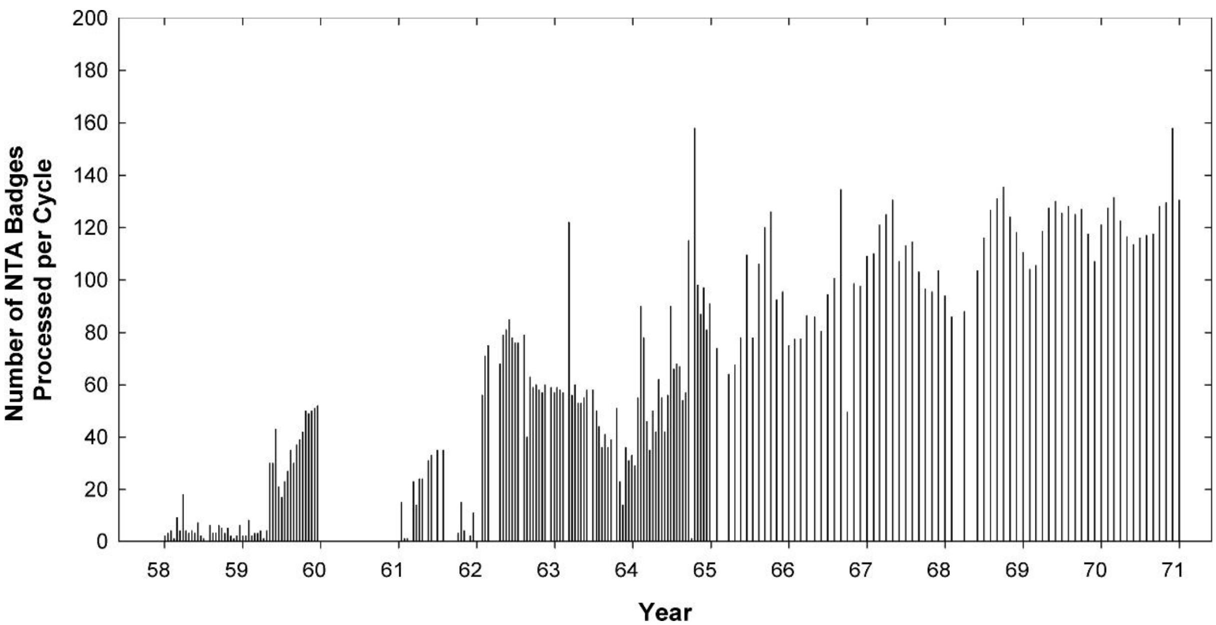


Figure 7-3: Number of Neutron Film Dosimeter Badges Processed per Cycle for the 200-F Area.

During this evaluation, NIOSH located and reviewed all of the NTA data for SRS from 1961 through 1971 when the TLND was implemented. These data are recorded on the NTA Process and Inventory Log Sheets and contain the number of tracks read per field of view and the conversion/calibration factor for each batch of NTA film used in an area. The inventory sheets are organized by area and contain individual monitoring results as well as summary results of the positive neutron doses. At the time of this writing, NIOSH has requested these records and they are currently undergoing classification review. NIOSH expects to receive these records by the end of November 2008.

In the past, NTA film data has been considered to be of limited value due to the relatively poor response (under-response) of the film to intermediate and low energy neutrons, generally below about 500 keV. This deficiency has been recognized industry-wide and the subject of continuing R&D with a continuing series of corrections applied to recorded NTA results. SRS, in addressing the neutron measurement technological difficulties, participated with other AEC/DOE laboratories and facilities in the solution of the issue both at SRS and industry-wide. As evidence of this continuing participation, an extensive study was performed and published that included complete neutron spectral measurements on all SRS workstation locations at which neutron exposures could occur. This study is available and can be used as another means of validation of the establishment of any necessary correction factors for the early NTA dose measurements (PNL-6301). However, the response or degree of under-estimation is highly dependent on the calibration source used. At some other facilities, a relatively low neutron energy source such as PuF_4 was used so that the under-response was generally less than a factor of two. However, at SRS the calibration source was initially a PoBe source and later changed to a PuBe source, both of which are high-energy neutron sources. As indicated in Table 6-3, prior to the change from NTA film to TLND, a comparison was conducted at the Plutonium Finishing Area. This comparison indicated that the NTA film under-responded by an average of 3.9. Thus, to accurately bound an individual worker's neutron dose at SRS, the NTA-measured neutron dose should be multiplied by 3.9. At the time the SRS TBD was written, the actual correction factor was not known because the results from this historical comparison study had not yet been located. As a result, a neutron-to-photon ratio was developed using TLND data (post-1971). This N:P ratio was back-extrapolated to cover the NTA time period. Now that more specific information related to the under-response is known, the current N:P ratio in the SRS TBD can be validated against the corrected NTA dose values as developed, using the recently recovered information.

7.2.2.3 Feasibility of Bounding Neutron Doses

At present, NIOSH has not found any evidence that the N:P ratios used in the SRS TBD are unreasonable, and therefore, cannot be used for dose reconstruction. However, given the new data that has been found as a result of this evaluation, NIOSH now has the ability to either validate or correct the current neutron-to-photon ratio used for SRS dose reconstructions.

For this validation, an N:P ratio will be developed based upon paired NTA and photon measurement data post-1962. The distribution of this ratio will be compared to the TBD value and NIOSH will either validate the current listed TBD N:P ratios or update them. The completion date for this validation is not known as of this writing because the information NIOSH reviewed in August 2008 has not yet been cleared and received. Upon receipt of this information, NIOSH expects the evaluation will take 3-4 months to complete. The NTA data in the claimant files will be corrected to total neutron dose from all energies based on the data in DP-MS-69-4 and the additional evaluations.

Since personnel neutron monitoring data are limited prior to 1962, paired neutron and photon dose rate measurements will be extracted from the Radiation Survey Log Sheets and used to develop the N:P ratio. This survey-based N:P ratio will be compared to both the current value in the TBD as well as the value developed using the corrected NTA monitoring data. The timeline for completion of this component is significantly longer than the NTA analysis because the individual survey forms that contain the paired measurements must first be tagged for scanning. The sheets must then be coded and the analysis conducted. For the analysis conducted at the Hanford Site, this effort took approximately six months after the initial data review/capture. Due to NIOSH and DOE resource limitations, NIOSH's tentative schedule is to review/capture this information in February 2009; thus, the validation using the survey-based data is expected to be complete in August 2009.

As noted during this evaluation, NIOSH did not uncover any evidence to indicate that the N:P ratios in the SRS TBD are not bounding. Through the limited reviews of the Radiation Survey Log Sheets, the N:P ratios appear to fall within the range specified in the TBD. NIOSH recognizes that further evaluation is now warranted given the extensive monitoring records now available.

As for the feasibility of neutron dose reconstruction, if neutron dose records exist for the individual worker, these should be used to reconstruct the neutron dose. For early time periods (NTA era), the measured neutron dose should be multiplied by a factor of 3.9. If neutron monitoring records do not exist, and it is established that the worker was potentially-exposed to neutrons in accordance with OCAS-TIB-007, then the neutron dose is calculated using the N:P ratios presented in the SRS TBD. The use of the N:P ratio is generally applied to pre-1971 data; however, for some workers, the ratio is also applicable post-1971. Effectively, for unmonitored construction trades workers, the co-worker photon dose multiplied by 1.4 (ORAUT-OTIB-0052) constitutes the base photon dose from which the N:P ratio is then applied to reconstruct the neutron dose.

In summary, the current neutron dose reconstruction methodology is considered to be sufficiently accurate for bounding neutron doses at SRS. During this evaluation, NIOSH has discovered additional neutron monitoring data that can be used to validate the current methodology or to revise the methodology, as appropriate. Regardless of whether the evaluation of the new data validates the current method or results in a new method, NIOSH believes that the extensive amount of data discovered is sufficient to support the establishment of a bounding neutron dose for the class under evaluation.

7.2.3 External Dose Reconstruction Feasibility Conclusion

The available monitoring data for photon and neutron doses, and the process and source term documentation, have been examined and found to be adequate for estimating bounding external doses for the class. The analysis of newly-acquired photon dose data not previously included in the HPAREH database shows that the current annual co-worker dose values are appropriate for dose reconstruction when individual monitoring results are not available. Neutron doses, for years when NTA film was used, can be reconstructed by adjusting (correcting) the NTA film dosimetry results or by applying the current neutron-to-photon ratio, or by developing a new neutron-to-photon ratio based either on NTA film data and/or on neutron radiation survey data. Therefore, the external dose to SRS construction workers may be bounded with sufficient accuracy based on the reported monitoring results and the assessment of the external monitoring results, as described in this section.

7.3 Evaluation of Petition Basis for SEC-00103

The following subsections evaluate the assertions made on behalf of and specific to petition SEC-00103 for the Savannah River Site.

7.3.1 Dose Reconstructions for Construction Workers

SEC-00103, Attachment 1, Page 43 of 91: *Construction Workers have missing external monitoring data, for all or parts of their employment periods, or "zero" results recorded for an entire year even though working in "radiation areas" or have been involved in radiation exposure incidents.*

This issue was addressed and resolved in Section 6.2. Based on the review of the annual dose distributions, and specific evaluation of the construction trades workers, NIOSH concludes that the HPAREH database used in ORAU-OTIB-0032 is sufficiently robust and can, therefore, be used to bound external photon and neutron doses.

7.3.2 Consideration of Construction Workers' Involvement in Incidents

SEC-00103, Attachment 2, Page 64 of 91: *NIOSH has not considered the involvement of construction workers in radiation incidents or accidents.*

NIOSH has considered each of these declared situations. In reviewing the SRS Site Incident database, NIOSH found that construction workers were recorded as being involved in about 25 percent of incidents. NIOSH further found that SRS HP provided monitoring for the particular situations (e.g., surface contamination monitoring, personnel contamination monitoring, follow-up contamination monitoring, personnel nasal smears, bioassay, and documentation for each involved worker). More discussion is presented in Section 5.2.3.

7.3.3 Exposure Patterns of Construction Workers

SEC-00103, Attachment 2, Page 64 of 91: *In regards to the SRS Site Profile: The Site Profile is skewed toward production workers who work in one facility, or area, for a long time and does not account for the very different exposure patterns of construction workers.*

Members of the proposed SEC Class were employees of DuPont and of private contractor organizations and were provided the same Health and Safety support as other operations and maintenance workers. When any worker entered a radiologically-controlled work area, SRS required dosimetry, protective clothing, and equipment as required for the job, and bioassay as needed. Based on the review described in Sections 5.2.3, 6.1, and 6.2, NIOSH concludes that the HPAREH database, used in conjunction ORAU-OTIB-0032, ORAU-OTIB-0052, and internal intake co-worker data, is sufficiently robust and can, therefore, be used to bound external and internal doses for the class under evaluation.

7.3.4 1990 Tiger Team Assessment

SEC-00103, Attachment 2, Page 64 of 91: *NIOSH has not considered deficiencies in the radiation monitoring programs such as those identified by the 1990 Tiger Team (DOE/EH-0133).*

NIOSH has reviewed the 1990 Tiger Team report as part of this SEC Evaluation Report. The Tiger Team identified 94 findings pertaining to radiation protection. A numerical summary of these items by category is shown in Table 7-5.

Table 7-5: Numerical Summary of Tiger Team Radiation Protections Findings

Finding Category	No. of Findings
Contamination Control	7
Radioactive Material Control	11
Inadequate Air Sampling / Monitoring	9
Effluent Environmental Monitoring	4
Instrument & Calibration	22
General Practice / Organization	11
ALARA	9
Training	9
Procedures & RWP	7
Dosimetry	5
Total	94

NIOSH reviewed the five findings pertaining to dosimetry:

- *Truck Drivers Wearing Dosimeter on Chest (source was behind driver)*

  NIOSH Response: While a posterior to anterior geometry can be used to account for dose, the external dose can be bounded using Section 7.2. A technical information bulletin will be published for use individual dose reconstructions to correct for the geometry.

- *TLD Storage Racks Inside RCA (entry violation)*

  NIOSH Response: In the case that dosimeters were stored inside an RCS, and workers were required to enter the RCA in order to get their dosimeter, this would result in a dosimeter-recorded reading higher than the actual radiation dose received by the individual. Therefore, this issue does not impact NIOSH's ability to bound the dose for the class under evaluation.

- *Inadequate Monitoring of Casual Visitors Inside RCA (they were not performing work)*

  NIOSH Response: Any person entering an RCA was required to wear a dosimeter, and if they were not, they were in direct violation of procedures. All visitors entering an RCA were issued visitor dosimeters. Based on the review and assessment performed by NIOSH for this report, this issue does not impact NIOSH's ability to bound the dose for the class under evaluation.

- *Workers Observed in RCA Not Wearing Proper Dosimetry Devices*

  NIOSH Response: Any person entering an RCA was required to wear a dosimeter, and if they were not, they were in direct violation of procedures. All visitors entering an RCA were issued visitor dosimeters. Based on the review and assessment performed by NIOSH for this report, this issue does not impact NIOSH's ability to bound the dose for the class under evaluation.

- *Inadequate Follow-up on Delinquent Bioassay*

  NIOSH Response: NIOSH is developing a co-worker model based on the claimant data in NOCTS to account for intakes of workers who may not have been monitored during periods of employment. A separate evaluation has been completed that establishes the principle that, under certain conditions, the data in NOCTS are representative of a site's population generally (ORAUT-OTIB-0075). Based on the review and assessment performed by NIOSH for this report, this issue does not impact NIOSH's ability to bound the dose for the class under evaluation.

7.3.5 Work in Non-radiological Areas

SEC-00103, Affidavits, Pages 8, 23, 24, 25, and 27 of 91: *Several affidavits claim instances of workers who were working in supposedly non-radiological areas, only to later find out that the area was contaminated and/or a radiation area.*

As noted in Section 7.3.2, NIOSH found in the SRS Site Incident database that construction workers were recorded as being involved in about 25% of incidents. NIOSH further found that SRS HP provided monitoring for the particular situations (e.g., surface contamination monitoring, personnel contamination monitoring, follow-up contamination monitoring, personnel nasal smears, and bioassay. In some instances, low levels of radiation were detected in the roped-off area but were not high enough to require personnel dosimetry according to standing procedure (i.e., anticipated exposures would be below permissible levels. From this analysis, NIOSH has established that areas where radiological work was conducted were monitored and that construction workers working in those areas were monitored. In the event it is determined that a specific worker was not monitored during a specific job, missed dose can be bounded using criteria provided in Sections 7.1 and 7.2. Based on the review and assessment performed by NIOSH for this report, this issue does not impact NIOSH's ability to bound the dose for the class under evaluation.

7.3.6 Cover-up of Incidents

SEC-00103, Affidavits, Page 15 of 91: *Statements were provided that alleged radiological incidents were covered up.*

NIOSH has found no information to substantiate that radiological incidents were covered up, or to substantiate that some were not. It is at least plausible to expect that some radiological incidents/accidents were not formally reported due to lack of radiological impact. In the example cited in the petition, the individual involved indicated that subsequent to the incident, he had left bioassay samples. As a result, due to the routine monitoring that was conducted, the potential for a serious overexposure to have gone unnoticed even with a cover-up is considered to be relatively small. Even if the potential was larger, NIOSH finds that bound doses derived using both routine personnel monitoring data available in NOCTS and HPAREH, and co-worker intakes (Section 7.1) would include doses received during potential "covered-up" incidents.

7.4 Other Potential SEC Issues Relevant to the Petition Identified During the Evaluation

ATTRIBUTION: Section 7.4 and its related subsections were completed by Ed Scalsky, Dade-Moeller, Inc.; Mike Mahathy, Oak Ridge Associated Universities; Eugene Potter, Mel Chew and Associates, Inc. These conclusions were peer-reviewed by the individuals listed on the cover page. The rationales for all conclusions in this document are explained in the associated text.

During the feasibility evaluation for SEC-00103, a number of issues were identified that needed further analysis and resolution. The issues and their current status are discussed in the following subsections.

7.4.1 Radionuclides from Special Campaigns Not Included in the TBD

Potential intakes of construction workers from exposures to, and subsequent intakes of, radionuclides handled or generated during Isotope Production Campaigns were not assessed.

NIOSH reviewed bioassay logbooks, comparing the dates given for analyses of specified radionuclides to the dates of SRS isotope campaigns. Results of that comparison are provided in Figure 7-1. Monitoring for exposure to all isotopes was performed except for curium-244 from 1959 through 1963. SRS Health Physics procedures for radiological monitoring applied to "All Departments and Construction" (DPSOP-40). Methods of reconstructing potential internal doses are discussed in Section 7.1

7.4.2 Tank Farm Exposures

During evaluation of the proposed class, NIOSH interviewed several former workers. A former worker stated that construction workers had performed frequent work at the Tank Farms, that the work was not monitored and that the source term had not been characterized.

NIOSH evaluated the source term of the tank farms, which consisted of tritium, plutonium, uranium (including enriched [1.1%] uranium), neptunium, americium, curium, thorium, polonium, and fission products. The activities ranged from high-level to low-level. From the examination of the SRS Site Incident database, NIOSH found that the Tank Farms were staffed by Health Physics and that there are monitoring data available for SRS incidents. Furthermore, NIOSH has established that SRS radiological workers were monitored for external and internal radiation exposure. For workers who should have been monitored but were not, both external and internal doses can be reconstructed using co-worker distributions for beta-gamma- emitting radionuclides, applying external correction factors given in ORAUT-OTIB-52 and applying intake correction factors established for construction workers based on stay times and ORAUT-OTIB-52 parameters. An internal co-worker intake distribution will be developed using all DOE-supplied bioassay data stored for SRS claimants in NOCTS. NIOSH has demonstrated that bioassay data reported for claimants of any particular DOE sites is representative of that site's population of bioassay data (ORAUT-OTIB-0075).

7.4.3 Forestry Workers

During the same interviews mentioned in Section 7.4.2, a former Forestry Service employee stated he worked at SRS where he tested and helped clear trees in the canyons and other areas; his office was in Building 760G. He stated that he had not been monitored.

NIOSH found that the worker was not monitored for external radiation exposure during his stay at SRS and employment with the Forestry Service. The worker did receive a whole-body count in 1992 while working as an escort. While the worker would not be a member of the SEC class under evaluation, NIOSH finds that doses to forestry workers can be bounded using environmental internal doses, and dose conversion factors given in ORAUT-TKBS-0003, pp. 188-192. Intakes were calculated with the assumption that all foodstuffs were grown in the areas of highest radionuclide concentrations and that the radionuclide content in the plant mass can be calculated by use of transfer factors given in PNNL-13421. The transfer factors provide a ratio of the plant mass to the soil concentration (dry mass-to-dry mass). External doses can be bounded using assumptions and environment dose conversion factors given in ORAUT-TKBS-0003, pp. 164-187.

7.4.4 Special Tritium Compounds

Does internal uptake of special tritium compounds result in a longer bodily retention of tritium than other forms of tritium?

NIOSH states that this issue does not prevent such doses from being bounded due to the large amount of tritium results available for SRS workers. The method for evaluating dose associated with special tritium compounds is documented in ORAUT-OTIB-0066, *Calculation of Dose from Intakes of Special Tritium Compounds*. This OTIB, coupled with the available SRS tritium monitoring data (as discussed in Section 7.0), supports NIOSH's ability to bound the dose associated with special tritium compounds for the class under evaluation.

7.4.5 Bounding of Doses from Intake of Thorium

Can potential internal doses received from exposure to thorium be bounded for construction workers with sufficient accuracy?

NIOSH has analyzed its ability to bound internal thorium dose for the class under evaluation. The results of that analysis are presented in Section 7.1.1.9. NIOSH concludes that, for the period from 1960 through 2003, available personnel monitoring data and dose reconstruction methods support the ability of NIOSH to bound the thorium internal dose for the class. However, NIOSH is continuing to evaluate the period from start-up through 1959, with that conclusion being RESERVED.

7.4.6 Completeness of the HPAREH Database

The Advisory Board contractor (SC&A) provided NIOSH with an inventory of supplemental records that may be beneficial in dose reconstruction. These records are not always provided in the claimant file submitted for dose reconstruction. No effort has been made to evaluate the completeness of the HPAREH file used in the development of the external co-worker model. The integrity of the HPAREH file for use in co-worker modeling is questionable given the absence of much of the data for workers terminating employment prior to 1979.

NIOSH has evaluated a comparison of doses in the Quarterly Summary Reports with those in HPAREH (see Section 6.0 of this report). As discussed, not all historical monitoring data was stored in the HPAREH database. The evaluation of the comparison was performed to examine the effect of the missing data on NIOSH's ability to bound the dose for the class under evaluation. NIOSH's review shows that the distribution of doses for construction workers in HPAREH is somewhat higher than the distribution of doses using Quarterly Summaries for most years. However, the construction worker Quarterly Summary distribution is bounded in all evaluated cases when compared to the HPAREH distribution for all monitored workers (which is the method applied in ORAUT-OTIB-0052 and the SRS external co-worker study). Based on the assessment of this approach, further refinement of the HPAREH distribution based on collection of additional data would only result in the reduction of the bounding values for the class. However, for the years where only a small percentage of HPAREH data exist (pre-1958 time frame), NIOSH will continue analysis of visitor logs and cards and refinements of the process to confirm this approach.

Based on the availability of SRS documentation, and NIOSH's review of the hardcopy records in comparison to the HAPREH database, there does not appear to be any significant administrative practice or data issue that would affect the integrity of the recorded SRS doses. The results of NIOSH's evaluation on this matter demonstrate that the use of HPAREH data is bounding for construction worker external photon doses.

7.4.7 Dosimetry Uncertainty

SC&A raised concerns about the beta/gamma correction factors. For the purpose of this evaluation, NIOSH states that only the SEC issues related to bounding the dose for the class must be considered in this report. As such, the evaluation of TBD correction factors or adjustments to recorded dose (which do not inhibit the ability of NIOSH to bound external doses) are generally considered Site Profile/TBD issues and not SEC issues (although some topics may be addressed in this evaluation for the purpose of completeness in the dosimetry accuracy discussions). In the interest of completeness, NIOSH is responding to the following uncertainty issues raised by SC&A.

1. *Calibration of dosimeters at 0 degrees is often not representative of incident angles encountered in the field and result in an underestimation of the true exposure that is being measured.*

An evaluation of several variables impacting the accuracy of the SRS Film Badge System was documented in a report prepared by Wright (Wright, 1958). This evaluation includes measured directional response characteristics of the SRS beta/photon film dosimeter throughout a full 360° rotation. Figure 7-4 (reproduced from the Wright report shows iso-exposure circles at 0.9, 0.8 and 0.7 in relation to the open-window exposure at 1.0 (i.e., 100%). The response to radium photon radiation by the dosimeter's shielded region (dashed lines) is effectively equivalent to 1.0 based on the calibration for a 150-degree arc for an anterior irradiation. Overall, the badge responded better than 0.9 and 0.8 over 90 % and 98 % of the viewing angles, respectively. Based on these figures, exposure from the posterior would result in an overresponse of the shielded region (i.e., greater than 1.0)

In Figure 7-4, the notable reduction in response shown at 0° or 180° for an anterior irradiation is minimized in the workplace because of movement of the dosimeter while being worn by the worker, and by the broad radiation fields characteristic of typical workplaces due to scatter and large and/or multiple sources of radiation. The response of the dosimeter's open-window region (solid line) is similar to the shielded response.

Effects of energy dependence on the directional response are presented in the International Agency for Research on Cancer (IARC) Intercomparison Study (Thierry, 2002) of ten widely-used dosimetry systems throughout the world. The IARC Study illustrated that film dosimeters typically overestimate the delivered dose for photon radiation greater than about 80 keV in rotational, isotropic, and anterior-posterior exposure geometries relative to the typical method of calibration using radium in an anterior-posterior geometry.

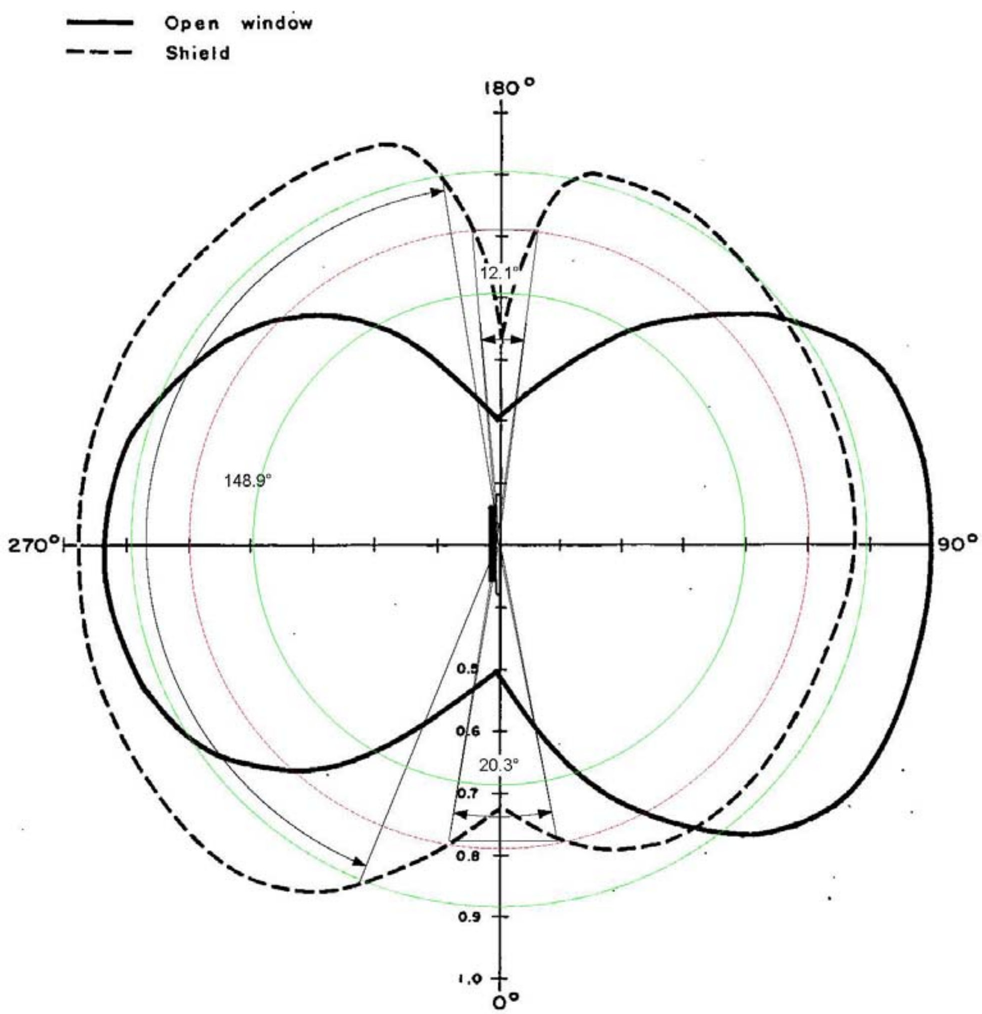


Figure 7-4: Directional Response of Savannah River Plant Film Badge (Gamma) (Wright, 1958)

2. *The on-phantom correction factor of 1.119 may be too low for photon energies between 30 and 250 keV.*

Reporting on the work of Goodwyn, Taylor provided the correction factors that were applied to bring the SRS dosimetry calibration into closer agreement with other sites and accepted standards (WSRC-RP-95-234). For the period prior to 1986, for all beta/photon dosimeters, a correction factor of 1.119 is applied; for 1986, a correction factor of 1.039 is applied; for the years 1987 to the present, no correction factor is required.

The original work by Goodwyn has been requested from SRS and is undergoing classification review. As reported in WSRC-RP-95-234, SRS staff provided in the Goodwyn reference an evaluation of the overall adjustments necessary to arrive at a consistent estimate of the deep dose equivalent (Hp(10)) as employed in dosimeter performance studies at that time using ANSI N13.11-1983. This apparently represented their judgment of complex parameters, as noted in the SRS TBD. SRS staff had knowledge of follow-up dose evaluations for incidents where workplace directional and spectral characteristics were considered, and comparisons of the dosimeter- and instrument-measured dose. The SRS TBD team concluded that (for dose reconstructions) an increase in the assigned dose by a factor of 1.119 should be employed in addition to other dose reconstruction considerations described in OCAS-IG-0001.

3. *The TBDs generic standard deviation value of 30% is likely to be low for film dosimeters prior to 1971. Early film dosimeters are likely to have a workplace standard deviation of at least 40%.*

Uncertainty in the dosimeter results is an important component of external dose reconstruction. A systematic personnel meter (beta-gamma film dosimeter) audit record was routinely produced (SRS, 1959). Visitor dosimeters were exposed to known doses of beta, gamma, and in some instances, X-ray radiation ranging from 0 to 300 mrem. The dosimeters were read and the results reported to Health Physics department management. A summary statement at the end of each report shows the average error for dose from gamma radiation, beta radiation, and sometimes the dose from beta plus gamma radiation. The April 3, 1961 audit marked the first use of the automatic densitometer in this testing program. Figure 7-5 shows the gamma dose results of this testing program based for the period March 19, 1959 through September 18, 1963. Additional records have been requested that may document results before and after this period. In general, the measured error seldom exceeded ±20%. The dosimeters tended to be biased high and the beta dose errors exceeded gamma dose errors. All of the measured errors tended to be smaller after introduction of the automatic densitometer.

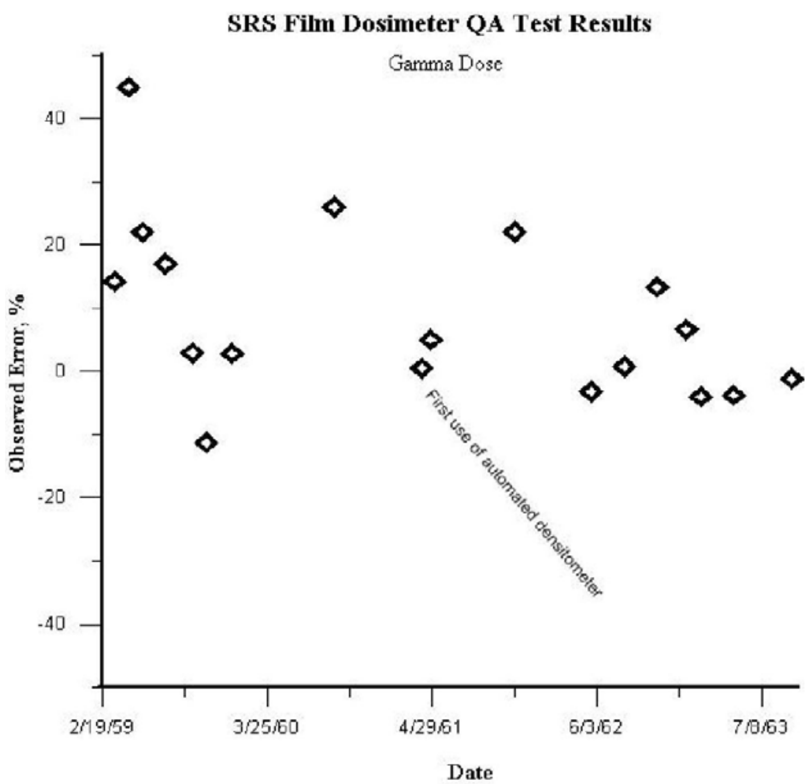


Figure 7-5: SRS Film Dosimeter Audit Results for March 1959 through September 1963

7.4.8 Early Construction Worker Monitoring Data

The TBD does not address the consistency of the SRS internal and external monitoring program for different operations and through time. This is especially important in the case of early workers who may not have had routine monitoring commensurate with their exposure.

NIOSH finds that the HPAREH data can be used to support bounding external doses for the class under evaluation. NIOSH can continue to use the external monitoring data recorded at SRS to develop and further analyze the statistical relationship between HPAREH and the SRS-supplied data. NIOSH has assessed the available bioassay data (in NOCTS personnel records and SRS logbooks) and is establishing an internal co-worker study based on the NOCTS data (to be supplemented by other bioassay data, as necessary). Based on NIOSH's reviews, the monitoring and radiological program information and data support that all workers entering a radiological area at the site were appropriately monitored (DPSOP-40). The only exception to this may have been when special exclusion zones were established, and when this occurred, the outer boundaries of the radiological areas were monitored.

7.4.9 Early Construction Worker Neutron Monitoring Data

NIOSH identified a lack of neutron monitoring data for all workers including construction workers in the time period from 1954 through 1960. Personnel monitoring for neutrons was only required when workers entered a radiation area where the neutron dose rate exceeded 1 mrem/hr. Thus, some unmonitored exposure to neutrons was likely. From the review of the SRS Incident database, NIOSH has determined that construction workers worked in areas of potential neutron exposures.

NIOSH has reviewed SRS radiation survey reports to determine if workplace neutron measurements were conducted in parallel with photon survey measurements. NIOSH has verified that this parallel logging did occur. Paired measurements were recorded on individual Radiation Survey Log Sheets and are available in the SRS records holdings. The combination of the corrected NTA data, along with this survey data, will be used by NIOSH to establish the neutron-to-photon ratio for use in bounding neutron doses from 1954 through 1960. This methodology is discussed in more detail in Section 7.2.

7.4.10 Neutron to Photon Ratio

SC&A stated that both the TLND recorded neutron doses between 1971 and 1995, as well as the pre-1971 neutron doses (derived by neutron-to-photon ratios) suffer from a high degree of uncertainty.

NIOSH has addressed this issue; see Sections 6.2, 7.2.2.1, 7.2.2.2 and 7.2.2.3 for discussion.

7.5 Summary of Feasibility Findings for Petition SEC-00103

This report evaluates the feasibility for completing dose reconstructions for employees at the Savannah River Site from January 1, 1950 through December 31, 2007. NIOSH found that the available monitoring records, process descriptions, and source term data available are sufficient to complete dose reconstructions for the class under evaluation.

Table 7-6 summarizes the results of the feasibility findings at SRS for each exposure source during the time period January 1, 1950 through December 31, 2007.

Table 7-6: Summary of Feasibility Findings for SEC-00103

January 1, 1950 through December 31, 2007

Source of Exposure	Reconstruction Feasible	Reconstruction Not Feasible
Internal[1]	**X**	
- U	X	
- Pu	X	
- Am	X	
- Np	X	
- Cm	X	
- H-3	X	
- Cf	X	
- Th	X (1953-1959 decision reserved)	
- Fission Products	X	
External	**X**	
- Gamma	X	
- Beta	X	
- Neutron	X	
- Occupational Medical X-ray	X	

[1] Internal includes an evaluation of urinalysis (in vitro), airborne dust, and lung (in vivo) data

As of October 1, 2008, a total of 1798 claims have been submitted to NIOSH for individuals who worked in construction at the Savannah River Site. Dose reconstructions have been completed for 1358 individuals (~75%).

8.0 Evaluation of Health Endangerment for Petition SEC-00103

The health endangerment determination for the class of employees covered by this evaluation report is governed by both EEOICPA and 42 C.F.R. § 83.13(c)(3). Under these requirements, if it is not feasible to bound with sufficient accuracy radiation doses for members of the class, NIOSH must also determine that there is a reasonable likelihood that such radiation doses may have endangered the health of members of the class. Section 83.13 requires NIOSH to assume that any duration of unprotected exposure may have endangered the health of members of a class when it has been established that the class may have been exposed to radiation during a discrete incident likely to have involved levels of exposure similarly high to those occurring during nuclear criticality incidents. If the occurrence of such an exceptionally high-level exposure has not been established, then NIOSH is required to specify that health was endangered for those workers who were employed for a number of work days aggregating at least 250 work days within the parameters established for the class or in combination with work days within the parameters established for one or more other classes of employees in the SEC.

After examination and cross-comparison of the available monitoring data, source term information, and process information, NIOSH's evaluation determined that it is feasible to bound radiation dose for members of the NIOSH-evaluated class with sufficient accuracy based on the sum of information available from available resources. Modification of the class definition regarding health endangerment and minimum required employment periods, therefore, is not required.

9.0 Class Conclusion for Petition SEC-00103

Based on its full research of the class under evaluation, NIOSH found no part of said class for which it cannot bound radiation doses with sufficient accuracy. This class includes all construction workers who worked in any area at the Savannah River Site during the period January 1, 1950 through December 31, 2007. However, NIOSH has reserved the feasibility determination for thorium exposures from January 1, 1950 through December 31, 1959; NIOSH is continuing to evaluate the thorium bounding approach for this time period.

NIOSH has carefully reviewed all material sent in by the petitioner, including the specific assertions stated in the petition, and has responded herein (see Section 7.4). NIOSH has also reviewed available technical resources and many other references, including the Site Research Database (SRDB), for information relevant to SEC-00103. In addition, NIOSH reviewed its NOCTS dose reconstruction database to identify EEOICPA-related dose reconstructions that might provide information relevant to the petition evaluation.

These actions are based on existing, approved NIOSH processes used in dose reconstruction for claims under EEOICPA. NIOSH's guiding principle in conducting these dose reconstructions is to ensure that the assumptions used are fair, consistent, and well-grounded in the best available science. Simultaneously, uncertainties in the science and data must be handled to the advantage, rather than to the detriment, of the petitioners. When adequate personal dose monitoring information is not available, or is very limited, NIOSH may use the highest reasonably possible radiation dose, based on reliable science, documented experience, and relevant data to determine the feasibility of reconstructing the dose of an SEC petition class. NIOSH contends that it has complied with these standards of performance in determining the feasibility or infeasibility of reconstructing dose for the class under evaluation.

10.0 References

42 C.F.R. pt. 81, *Guidelines for Determining the Probability of Causation Under the Energy Employees Occupational Illness Compensation Program Act of 2000;* Final Rule, Federal Register/Vol. 67, No. 85/Thursday, p 22,296; May 2, 2002; SRDB Ref ID: 19391

42 C.F.R. pt. 82, *Methods for Radiation Dose Reconstruction Under the Energy Employees Occupational Illness Compensation Program Act of 2000*; Final Rule; May 2, 2002; SRDB Ref ID: 19392

42 C.F.R. pt. 83, *Procedures for Designating Classes of Employees as Members of the Special Exposure Cohort Under the Energy Employees Occupational Illness Compensation Program Act of 2000*; Final Rule; May 28, 2004; SRDB Ref ID: 22001

42 U.S.C. §§ 7384-7385 (EEOICPA), *Energy Employees Occupational Illness Compensation Program Act of 2000*, as amended

DNA-1251-1-EX, *Compilation of Local Fallout Data from Test Detonations 1945-1962 Extracted from DASA 1251*, Volume 1, H. A. Hawthorne; May 1, 1979; SRDB Ref ID: 8859

DOE/DP-0137, *Plutonium: The First 50 Years – United States Plutonium Production, Acquisition, and Utilization from 1944-1994*, February 1996; SRDB Ref ID: 12292

DOE/EA-0948, *Environmental Assessment Operation of the HB-Line Facility and Frame Waste Recovery Process for Production of PU-238 Oxide at the Savannah River Site,* U.S. Department of Energy, Savannah River Operations Office; April 1995; SRDB Ref ID: 48942

DP-1222, *Separation of 210Po from Bismuth by Liquid Metal Distillation*, R. F. Bradley, L. F. Landon, G. W. Gibson, E. I. du Pont de Nemours and Company, June 1970; SRDB Ref ID: 48630

DP-MS-69-4, *A Thermoluminescent Personnel Neutron Dosimeter,* A. Korba, J. E. Hoy, E. I. du Pont de Nemours and Company, October, 9, 1969; SRDB Ref ID: 48550

DPSOP-40, *Savannah River Plant Radiation and Contamination Control – History Copy*, Vol. I and II; E. I. du Pont de Nemours and Company Explosives Department, date unknown; SRDB Ref ID: Not yet in SRDB

DPSOP-47, *Westinghouse Savannah River Company Radiobioassay Laboratory Procedures*, Rev. 14, SRP Health Protection Department; July 1987; SRDB Ref ID: 45959

DPSOX-254, *Test Authorization 5-1: Film Badge Issuance*, C. M. Patterson; November 23, 1953; SRDB Ref ID: 49202

DPSP-53-1-1, *Monthly Progress Report for January 1953*, E. I. du Pont de Nemours and Company, February 6, 1953; SRDB Ref ID: 11419

DPSP 54-1-10, *Monthly Progress Report for October 1954*, E. I. du Pont de Nemours and Company, November 16, 1954; SRDB Ref ID: 11458

DPSP 54-1-11, *Monthly Progress Report for November 1954*, E. I. du Pont de Nemours and Company, December 20, 1954; SRDB Ref ID: 11459

DPSP 55-1-8-DV, *Monthly Progress Report, August 1955*, E. I. du Pont de Nemours and Company, September 19, 1955; SRDB Ref ID: 11473

DPSP 55-1-9-DV, *Monthly Progress Report, September 1955,* E. I. du Pont de Nemours and Company, October 21, 1955, SRDB Ref ID: 11476

DPSP-55-454-2, *Savannah River Plant History, Separations Areas, July 1954 through December 1972*, E. I. du Pont de Nemours and Company, April 1973; SRDB Ref ID: 48560

DPSP-55-454-3, *Savannah River Plant History, Raw Materials Area, July 1954 through December 1972*, E. I. du Pont de Nemours and Company, February 1974; SRDB Ref ID: 48563

DPSP 56-1-8, *Monthly Progress Report, August 1956*, E. I. du Pont de Nemours and Company, September 18, 1956; SRDB Ref ID: 12057

DPSP 56-1-9, *Monthly Progress Report, September 1956*, E. I. du Pont de Nemours and Company, October 18, 1956; SRDB Ref ID: 16859

DPSP 57-1-1, *Monthly Progress Report, January 1957*, E. I. du Pont de Nemours and Company, February 18, 1957; SRDB Ref ID: 11575

DPSP 57-1-2, *Monthly Progress Report, February 1957,* E. I. du Pont de Nemours and Company, March 20, 1957; SRDB Ref ID: 11576

DPSP 57-1-7, *Monthly Progress Report, July 1957*, E. I. du Pont de Nemours and Company, August 19, 1957; SRDB Ref ID: 11601

DPSP 57-1-8, *Monthly Progress Report, August 1957*, E. I. du Pont de Nemours and Company, September 19, 1957; SRDB Ref ID: 11602

DPSP 57-1-9, *Monthly Progress Report, September 1957*, E. I. du Pont de Nemours and Company, October 18, 1957; SRDB Ref ID: 11603

DPSP 57-1-11-DV, *Monthly Progress Report, November 1957*, E. I. du Pont de Nemours and Company, December 18, 1957; SRDB Ref ID: 11605

DPSP 58-1-12-DV, *Monthly Progress Report, December 1958*, E. I. du Pont de Nemours and Company, January 25, 1959; SRDB Ref ID: 11615

DPSP 59-1-6, *Progress Report June 1959*, E. I. du Pont de Nemours and Company, July 21, 1959; SRDB Ref ID: 11628

DPSP 60-1-10-DV, Progress Report, October 1960, E. I. du Pont de Nemours and Company, November 17, 1960; SRDB Ref ID: 11786

DPSP-61-1-2, *Progress Report – February 1961*, H. L. Butler, E. I. du Pont de Nemours & Co., March 16, 1961; SRDB Ref ID: 12453

DPSP 61-1-10-DV, *Progress Report, October 1961*, E. I. du Pont de Nemours and Company, November 20, 1961; SRDB Ref ID: 11796

DPSP-67-1281, *Polonium Production in Curium-II*, P. L. Gray, June 30, 1967; SRDB Ref ID: 49819

DPSPU-58-030-011, *Rapid Determination of Mixed Beta-Gamma Radionuclides in Urine*, A. L. Boni, E. I. du Pont de Nemours and Company; SRDB Ref ID: 49225

DPSPU-58-030-014, *Radiation Protection Considerations in a Heavy Water Moderated Reactor*, Savannah River Plant; E. I. du Pont de Nemours & Co.; September 1958; SRDB Ref ID: 48566

DPSPU-60-11-7, *Sensitive Methods for Determination of Tritium in Water and Urine*, F. E. Butler, June 1960; SRDB Ref ID: 49513

DPSPU 62-30-5, *Control of Tritium Health Hazards at the Savannah River Plant*, W. C. Reinig, E. L. Albeneslus, Savannah River Plant, January 1962; SRDB Ref ID: 48558

DPSPU-62-30-25A, *Control of Tritium Hazards in Heavy-Water-Moderated Reactors*, H. L. Butler, C. M. Patterson, September 1962; SRDB Ref ID: 49903

DPSPU-63-30-037, *Assessment of Tritium in Production Workers*, F. E. Butler, Savannah River Plant, March 1964; SRDB Ref ID: 49905

DPSPU 63-30-30, *Determination of Actinides in Urine by Liquid Ion Exchange*, F. E. Butler, E. I. du Pont de Nemours and Company, August 1963; SRDB Ref ID: 49584

DPSPU 64-30-18, *Separation of Actinides by Liquid Ion Exchange, Determination of U and Am-Cm in Urine*, F. E. Butler, September 1964; SRDB Ref ID: 49523

DPSPU-75-30-7, *Radiation Exposure Records Management*, H. P. Boitar, Du Pont, December 1975; SRDB Ref ID: 8811

NSEC008-P006, *Startup to December 14, 1954*, author unknown, date unknown; SRDB Ref ID: 48929

OCAS-IG-001, *External Dos Reconstruction Implementation Guideline*, Rev. 3; National Institute for Occupational Safety and Health (NIOSH); Cincinnati, Ohio; November 21, 2007; SRDB Ref ID: 38864

OCAS-PER-030, *Savannah River Site TBD Revisions,*; Rev. 0; National Institute for Occupational Safety and Health (NIOSH); Cincinnati, Ohio; December 12, 2007; SRDB Ref ID: 38872

OCAS-PR-004, *Internal Procedures for the Evaluation of Special Exposure Cohort Petitions*, Rev. 0, National Institute for Occupational Safety and Health (NIOSH); Cincinnati, Ohio; September 23, 2004; SRDB Ref ID: 32022

OCAS-TIB-006, *Interpretation of External Dosimetry Records at the Savannah River Site (SRS)*, Rev. 2; National Institute for Occupational Safety and Health (NIOSH); Cincinnati, Ohio; October 4, 2007; SRDB Ref ID: 35409

OCAS-TIB-007, *Neutron Exposures at the Savannah River Site*, Rev. 01; National Institute for Occupational Safety and Health (NIOSH); Cincinnati, Ohio; October 15 2007; SRDB Ref ID: 35675

ORAUT-OTIB-0001, *Technical Information Bulletin: Maximum Internal Dose Estimates for Savannah River Site Claims*, Rev 00; Oak Ridge Associated Universities; July 15, 2003; SRDB Ref ID: 19407

ORAUT-OTIB-0006, *Dose Reconstruction from Occupationally Related Diagnostic X-Ray Procedures*, Rev. 03 PC-1; Oak Ridge Associated Universities; December 21, 2005; SRDB Ref ID: 20220

ORAUT-OTIB-0014, *Technical Information Bulletin: Assignment of Environmental Internal Doses for Employees Not Exposed to Airborne Radionuclides in the Workplace*, Rev 00; Oak Ridge Associated Universities; June 22, 2004; SRDB Ref ID: 19432

ORAUT-OTIB-0017, *Interpretation of Dosimetry Data for Assignment of Shallow Dose*, Rev. 01; Oak Ridge Associated Universities; October 11, 2005; SRDB Ref ID: 19434

ORAUT-OTIB-0018, *Technical Information Bulletin: Internal Dose Overestimates for Facilities with Air Sampling Programs, Rev 01*, Oak Ridge Associated Universities; August 9, 2005; SRDB Reference ID 19436

ORAUT-OTIB-0020, *Use of Coworker Dosimetry Data for External Dose Assignment*, Rev. 01; Oak Ridge Associated Universities; October 7, 2005; SRDB Ref ID: 19440

ORAUT-OTIB-0032, *External Coworker Dosimetry Data for the Savannah River Site*, Rev. 00, PC-1; Oak Ridge Associated Universities; November 7, 2006; SRDB Ref ID: 29964

ORAUT-OTIB-0052, *Parameters to Consider When Processing Claims for Construction Trade Workers*, Oak Ridge Associated Universities; Rev. 00 PC-1, January 16, 2007; SRDB Ref ID: 29978

ORAUT-OTIB-0060, *Technical Information Bulletin: Internal Dose Reconstruction*, Rev 00, Oak Ridge Associated Universities; February 6, 2007; SRDB Ref ID: 29984

ORAUT-PROC-0061, *Occupational X-Ray Dose Reconstruction for DOE Sites*, Rev. 01; Oak Ridge Associated Universities; July 21, 2006; SRDB Ref ID: 29987

ORAUT-OTIB-0066, *Calculation of Dose from Intakes of Special Tritium Compounds,* Rev. 00; Oak Ridge Associated Universities; April 26, 2007; SRDB Ref ID: 31421

ORAUT-OTIB-0075, *Use of Claimant Data Set for Coworker Modeling*, Oak Ridge Associated Universities; No date yet; SRDB Ref ID: to be published in 2008

ORAUT-TKBS-0003, *Savannah River Site,* Rev. 02; Oak Ridge Associated Universities; April 5, 2005; SRDB Ref ID: 20176

PNL-6301, 1987, *Neutron Dose and Energy Spectra Measurements at Savannah River Plant,* L. W. Brackenbush, et al; Pacific Northwest Laboratory; August 1987; SRDB Ref ID: 263

PNNL-13421, *A Compendium of Transfer Factors for Agricultural and Animal Products (June 2003*), K. Rhoades, B. A. Napier, D. L. Strenge, L. H. Staven, PNNL, June 2003; SRDB Ref ID: 23593

QH000-0385-92-002-P003, *SRS Dosimetry Quarterly Report for Period Ending September 1970*, Health Physics Department, October 21, 1970, SRDB Ref ID: 50075

QH230-0521-93-004-P006, *SRS Dosimetry Quarterly Report for Third Quarter 1965*, October 8, 1965; SRDB Ref ID: 50458

WSRC-IM-90-139, *Internal Dosimetry Technical Basis Manual*, Rev 1; December 20, 1990; SRDB Ref ID: 11266

WSRC-IM-94-39, *History of 232-F, Tritium Extraction Processing (U)*, G. W. Blackburn, August 1994; SRDB Ref ID: 12094

WSRC-MS-2000-00061, *Savannah River Site Fiftieth Anniversary, 50 Years of Excellence in Science and Engineering at the Savannah River Site*, J. W. Joseph, C. J. Banick, May 17, 2000; SRDB Ref ID: 45046

WSRC-RP-95-234, *A History of Personnel Radiation Dosimetry at the Savannah River Site (U)*, G. A. Taylor, K. W. Crase, T. R. LaBone, W. H. Wilkie, May 1995; SRDB Ref ID: 12098

WSRC-SA-11, *Receiving Basin for Offsite Fuels and the Resin Regeneration Facility Safety Analysis Report, Executive Summary, Chapters 1, 2, 3, 4, 5, 6, 7, 8, 9, 10, 11, 12, 13, 14, 15, 16, 17,* C. B. Shedrow, August 1997; SRDB Ref ID: 48970

Arcano, 1994, *Review of Facilities and Unirradiated Fuel Storage at the Reactor Materials Area of the Savannah River Site, March – April 1994*, J. T. Arcano Jr., Defense Nuclear Facilities Safety Board, May 19, 1994; SRDB Ref ID: 44906

Barg, 1996, *Litco International Report: Technical Basis for the SMC Bioassay Program*, Rev. 1, D. C. Barg, B. L. Anderson, January 16, 1996; SRDB Ref ID: 27055

Bio-Assay Procedure, 1958, *Bio-Assay Procedure for Tritium and Thorium,* Ellett, J. D., February 13, 1958; SRDB Ref ID: 48462

Bio-Assay Samples, 1957, *Bio-Assay Samples 100-K Area Shutdown*, J. A. Harper, September 5, 1957; SRDB Ref ID: 245

Bioassay Logbook, 1954-57, Bioassay *Fission Products Controls Record Book, 12-20-1954 Thru 1-4-1957*; no attribution; 1954-1957; SRDB Ref ID: 48920

Bebbington, 1990, *History of Du Pont at the Savannah River Plant*, W. P. Bebbington, 1990; SRDB Ref ID: 11249

Branche, 2008, Letter to Petitioner from NIOSH Office of the Director concerning Administrative Review Panel Findings, C.M. Branche, August 14, 2008, OSA Ref ID: 106860

Cardarelli, date unknown, Power Point Presentation: *Study of Radiation Doses-Potential Consequences of Excluding Work-Required X-Ray Exposures When Computing Cumulative* J. Cardarelli, date unknown; SRDB Ref ID: 11648

Clemons, 1993, *SRS ALARA Assessment June 28 – July 2, 1993*, L. Clemons and R. Zavadoski, Defense Nuclear Safety Board, September 21, 1993; SRDB Ref ID: 48950

DOE, 2008, *Environmental Management Savannah River Site,* U.S. Department of Energy, Environmental Management website; http://www.em.doe.gov/bemr/bemrsites/sars.aspx?182,128; 9/19/2008; SRDB Ref ID: 48954

Ellett, 1958, *Bio-Assay Procedure for Tritium and Thorium*, J. D. Ellett and C. M. Patterson; February 13, 1958; SRDB Ref ID: 48462

Exposure Evaluation, 1964, *Exposure Evaluation and Dosimetry Program Review*, H. L. Butler, January 23, 1964; SRDB Ref ID: 48904

FAS, 1998, *Heavy Water Production*, Federation of American Scientists; October 21, 1998; SRDB Ref ID: 48957

Fix, 1994, *An Assessment of Bias and Uncertainty in Recorded Dose from External Sources of Radiation for Workers at the Hanford Site*, J. J. Fix, E. S. Gilbert, W. V. Baumgartner; August 1994; SRDB Ref ID: 305

Fix, 2006, *Record of Phone Discussion with Ken Crase – Summary of Discussion of Selected Cohen & Associates SRS Site Profile Comments*, J. Fix; August 15, 2006; SRDB Ref ID: 27056

Hall, 1967, *A Comparison of Lithium Fluoride and Film for Personnel Dosimetry*, R. M. Hall, C. N. Wright; February 1, 1967; SRDB Ref ID: 11274

Health Physics, 1954, *Health Physics Area Survey Report April 1954*, C.M. Patterson, May 05, 1954; SRDB Ref ID: 49710

Health Physics, 1957, *Health Physics Bio-Assay Program 100-L Extended Shutdown*, J. L. Elrod, J. W. Harper, August 26, 1957; SRDB Ref ID:49759

Health Physics Aspects, 1958, *Health Physics Aspects of Reactor Operations 1956 versus 1957,* H. A. McClearen, January 17, 1958; SRDB Ref ID: 49506

Huntoon, 1954, *Test Authorization 3-138: AIS1 Canning Thorium Slugs*, R. T. Huntoon, J. R. Driear; August 27, 1954; SRDB Ref ID: 49207
Logbook P006, 1956, *Thorium Record Book, 2-28-1956 Thru 11-7-1956*, Author unknown; Savannah River Site; SRDB Ref ID: 49573

Morris, 2008, *SRS Thorium Processing Timeline to Support SRS SEC Evaluation Report*, Robert Morris, September 26, 2008; SRDB Ref ID: Not yet in SRDB

Personal Communication, 2008a, *Personal Communication with current worker, Summary of Discussion of Selected Cohen & Associates SRS Site Profile Comments;* Telephone Interview by ORAU Team; August 14, 2006; SRDB Ref ID: 27056

Personal Communication, 2008b, *Personal Communication with former worker, Use of NTA Neutron Personnel Dosimetry in 1955-1968 at SRS*; Telephone Interview by ORAU Team; March 9, 2008; SRDB Ref ID: 48459

Personal Communication, 2008c, *Personal Communication with current worker,* S*RS Incident Database, Special Hazards Investigations Database, HPAREH Database and Intake Registry;* Telephone Interview by ORAU Team; April 3, 2008; SRDB Ref ID: 45062

Personal Communication, 2008d, *Personal Communication with engineer concerning thorium,* Telephone Interview by ORAU Team; August 12, 2008; SRDB Ref ID: Not yet released by DOE

Personal Communication, 2008e, *Personal Communication with metallurgical lab technician,* Telephone Interview by ORAU Team; August 11, 2008; SRDB Ref ID: Not yet released by DOE

Personal Communication, 2008f, *Personal Communication with electrician,* Telephone Interview ORAU Team; August 11, 2008; SRDB Ref ID: Not yet released by DOE

Personal Communication, 2008g, *Personal Communication with former Health Physics manager,* Telephone Interview ORAU Team; August 11, 2008; SRDB Ref ID: Not yet released by DOE

Peterson, 2002, *Tritium Doses from Chronic Atmospheric Releases: A New Approach Proposed for Regulatory Compliance*, Volume 82, S. R. Peterson and P. A. Davis; February 2002; SRDB Ref ID: 46071

Santos, 2007, *Decommissioning of the 247-F Fuel Manufacturing Facility at the Savannah River Site*, J. K. Santos and S. M. Chostner; SRDB Ref ID: 48948

SEC, 2008a, *SEC Worker Outreach Meeting for the Savannah River Site*, National Institute for Occupational Safety and Health; May 22, 2008, 1 PM; SRDB Ref ID: Released by DOE; not yet in SRDB

SEC, 2008b, *SEC Worker Outreach Meeting for the Savannah River Site*, National Institute for Occupational Safety and Health; May 22, 2008, 6 PM; SRDB Ref ID: Not yet released by DOE

Singh, 2007, *Excerpts and Notes from the Site Incident Databank at SRS,* L. P. Singh, DOE, April 11, 2007; SRDB Ref ID: 32167

SRS, 1959, *SRS Personnel Meter Audit Record 1959-1963*, author unknown, March 19, 1959; SRDB Ref ID: 11291

SRS, 2008, *SRS Historical Highlights of the Savannah River Site,* SRS, May 20, 2008; SRDB Ref ID: 48960

SRS at Fifty, 2002, *Savannah River Site at Fifty – Reactor On*, M. B. Reed, M. T. Swanson, S. Gaither, J. W. Joseph, W. R. Harvey; U.S. Department of Energy and the Savannah River Site Operations Office; 2002; SRDB Ref ID: 24750

SRS Defense, 2007, *SRS Facts – Defense Waste Processing Facility*, December 2007, Washington Savannah River Company; SRDB Ref ID: 48961

SRS Dosimetry, 1959, *Dosimetry Special Hazards Incident Investigations 001-099*, April 29, 1959; SRDB Ref ID: 45086

SRS Dosimetry, 1962, *Dosimetry Special Hazards Incident Investigations 100-199*, Endorf, F. H., July 3, 1962; SRDB Ref ID: 45093

SRS Dosimetry, 1974a, *Dosimetry Special Hazards Incident Investigations 200-299*, Endorf, F. H., April 24, 1974; SRDB Ref ID: 45094

SRS Dosimetry, 1974b, *Dosimetry Special Hazards Incident Investigations 300-399*, Endorf, F. H., July 07, 1974; SRDB Ref ID: 45096

SRS Dosimetry, 1989, *Dosimetry Special Hazards Incident Investigations 400-494*, December 23, 1989; SRDB Ref ID: 45098

SRS F, 2008, *SRS Facts – F Canyon,* Washington Savannah River Company, January 2008; SRDB Ref ID: 48962

SRS FB, 2007, *SRS Facts – FB Line*, Washington Savannah River Company, December 2007; SRDB Ref ID: 48963

SRS H, 2007, *SRS Facts – H Area Completion*, June 5, 2007; SRDB Ref ID: 48964

SRS Highlights, 2000, *SRS Historical Highlights,* DOE Savannah River Operations Office, September 22, 2000; SRDB Ref ID: 11246

SRS Intake, 2004, *SRS Intake Database*, May 12, 2004; SRDB Ref ID: 45076

SRS Saltstone, 2007, *SRS Facts – Saltstone Facility*, Washington Savannah River Company, December 2007; SRDB Ref ID: 48482

SRS Spent, 2008, *SRS Facts – Spent Nuclear Fuel at SRS*, Washington Savannah River Company, January 2008; SRDB Ref ID: 49483

SRS Tritium, 2008, *SRS Facts - SRS Tritium Facilities*, Washington Savannah River Company, January 2008; SRDB Ref ID: 48969

Summary of Tritium, 1954, *Summary of Tritium Contamination May 12-July 19, 1954*, C. M. Patterson, July 21, 1954; SRDB Ref ID: 49701

Thierry, 2002, *Study of a Selection of 10 Historical Types of Dosimeter: Variation of the Response to Hp (10) with Photon Energy and Geometry of Exposure*, I. Thierry, et al; April 19, 2002; SRDB Ref ID: 11296

Washington Savannah River Company, 2008, *SRS Facts – H Canyon*, Washington Savannah River Company, January, 2008; SRDB Ref ID: 48965

Wright, 1958, *An Evaluation of the Savannah River Plant Film Badge System*, C. N. Wright; February 6, 1958; SRDB Ref ID: 48461

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Attachment 1: Savannah River Site Events Chronology

Table A-1: Savannah River Site Events Chronology		
Dates	**Areas**	**Activities**
February 1, 1951	400-D; K, C, L, P, R Reactors; 200-F and 200-H; 300-M	Construction begins on on-site railroads; the Heavy Water Rework Facility (400-D Area) and its supporting coal-fired steam/power plant; the K, C L, P, and R Reactors (100 Areas); F and H Canyons (300 Areas); the Fuel and Target Fabrication Facility (300-M); and the CMX and TNX technical development facilities.
June 1952	320-M	Normal assay uranium fuel slug cylinders manufactured at off-site locations begin to arrive at SRS.
July 1952	M	Several production support activities begin in the Fuel Target Manufacturing and Raw Materials Area.
August 8, 1952	400-D	D_2O separation test operations begin at the Heavy Water Rework Facility.
September 1952	305-M	Graphite Test Pile reactor achieves first criticality.
October 1952	400-D	Full-scale operation of the D-400 Heavy Water Rework Facility begins.
December 1952	305-M	Heavy-water-moderated SRS prototype reactor, the "Process Development Pile" (PDP) achieves full operations status.
January 1953	320-M	320-M fuel fabrication facility begins full-scale operations. First batch of SRS-produced "Mark I" finished fuel slugs are successfully tested at the end of the month.
May 1953	400-D	All heavy water production processes are operating at full capacity.
July 1953	320-M	First batch of SRS-produced control rods are successfully tested.
December 2, 1953	105 Areas	AEC authorizes start-up of L Reactor using enriched uranium fuel, and P, K, C Reactors to be loaded "for maximum plutonium production, with excess reactivity used for tritium production."
December 28, 1953	105-R	R Reactor achieves first criticality using Mark I natural uranium fuel.
February 20, 1954	105-P	P Reactor achieves first criticality.
June 1954	105-R	First irradiated fuel containing plutonium is withdrawn from R Reactor.
August 1954	313-M and 320-M	R&D and small-scale production activities began to can thorium metal slugs for inclusion in SRS reactor fuel.
August 11, 1954	105-L,	L Reactors achieves first criticality.
August 14, 1954	105-K	K Reactor achieves first criticality.
November 1954	221-F	Radioisotope separation work begins in F Canyon, A-Line Facility, and 221-F B-Line Facility.
November 1954	200-F	First delivery of plutonium product to AEC.
December 1954	200-F	HLW and LLW Tanks 1-7 placed in service.
December 1954	E	76 acres between F and H Areas designated as SRS solid waste burial ground.
February 1955	313-M and 320-M	SRS begins inspection and acceptance testing of thorium metal slugs canned by offsite vendors.
March 28, 1955	105-C	C Reactor achieves first criticality. All five SRS production reactors are now in operation.

Table A-1: Savannah River Site Events Chronology		
Dates	**Areas**	**Activities**
June 30, 1955	F-Area	First high-level waste tank is filled.
July 1955	221-H	Purex radioisotope separation work begins in H Canyon.
October 1955	200-F	First tritium recovered from irradiated lithium-aluminum alloy targets.
November 1955	217-F	First delivery of tritium gas product to AEC.
December 1955	F- and H-Areas	Waste tanks 2-F, 9-H, 10-H, and 11-H put into service storing waste from Purex process.
June 1956	232-H	Tritium Facility goes into operation.
November 1956	F- and H-Areas	Waste tanks 3-F, 8-F, 12-H, and 13-H are filled. Four additional waste tanks placed in service in H Area.
March 1957	F	F Area operations suspended to allow a major upgrade of equipment to achieve greater capacity, including construction of the JB Line on the F Canyon roof.
November 1957	H	14-H Waste Tank placed into service.
December 1957	D	411-D Building placed in standby and half of the DW Plant converted to rework tritiated moderator.
1958	100	P and R Reactor (Par) Pond, needed to provide cooling water for the P and R Reactors, is completed.
1958	K-100	SRS initiates production of small quantities of Pu-238 for use by NASA as heat sources.
1958	D	Building 413-D (D_2O extraction) and Plant E placed on standby.
1959	F	Purex operations resume in F Canyon.
1959	F	Waste tanks 5-F and 18-F receive high- and low-level wastes, respectively. Tanks 17-F and 20-F are constructed.
1960	H	SRS initiates construction of the Receiving Basin for Offsite Fuels (RBOF)
1960	F	Evaporation of waste started in F Area.
1960	H and F	15-H and 20-F waste tanks put into service.
1960	300	Hot press bonding slug production begins.
1961	F and H	F and H Canyon facilities begin alternating work schedules.
1961	F	Neptunium reprocessing begins in Building 235-F.
1961	H	Off-site tritium containers begin to arrive at Building 234-H.
1961	F	Wastes are placed in Tanks 4-F, 17-F, 19-F, and 21-F.
1961	H	Tanks 21-H and 24-H are placed into service.
1962	All	SRS begins installing air filters on some process emission sources.
1962	B	“Hector” - Heavy Water Components Test Reactor (770-U) begins operations.
1962	H	Tank 11-H receives HM high-level waste.
1963	H	Tank 12-H receives HM high-level waste.
1963	F	Building 235-F switched from neptunium to Cm-244 fabrication.
June 1964	R	R Reactor shuts down.
1964	B	Heavy Water Components Test Reactor is shut down.
1964	H	Receiving Basin for Off-Site Fuels (RBOF) receives first shipment of spent fuel assemblies.
1964	F	Waste Tank 6-F receives Purex high-level waste.
1964	H	Receiving Basin for Off-Site Fuels (RBOF) sends waste to Tank 23-H.

Table A-1: Savannah River Site Events Chronology		
Dates	**Areas**	**Activities**
1964	H	"Interim 23" U-233 separation from thorium fuel campaign begins.
1965	H	RBOF ships first off-site fuel to H-Canyon for processing.
1965	F and H	Simultaneous operations resume in both canyons.
September 1965	H	"Thorex" separation process to extract U-233 and thorium from thorium fuels begins.
1967	F and H	Waste transfer line connecting H and F Areas completed.
1968	L	L Reactor is shut down for design upgrades.
1968	C	C Reactor is shut down for repairs to tank wall.
1969	H	Tritium Facility begins recycling reservoirs.
November 1969	H	Final "Thorex" campaign is completed.
1970	F	Construction begins on Multi-Purpose Processing Facility (MPPF).
1971	D	Standby equipment in Buildings 411-D, 413-D, and E Plant is dismantled and sold for scrap.
1971	H	Waste Tanks 29-H and 32-H begin receiving wastes.
1972	H	Tank 31-H begins receiving waste concentrate.
1972	F Canyon	MPPF Building completed and put on standby.
1973	All	SRS adopts use of thermoluminescent dosimetry.
1973	F	Tank 34-F begins receiving waste concentrate.
1974	H	H Area processes low-enriched off-site fuel.
April 1974	F	Tank 33-F placed in service.
February 1975	F	Fire and explosion in A-Line necessitates shutdown of F Canyon
August 1975	F	F Canyon operations restart.
1977	F	Pu-238 production operations begin in the SRS Plutonium Fuel Form Facility (PUFF)
1978	F	Operations begin at the $^{238}PuO_2$ Experimental Facility (PEF).
1979	M	Process Development Pile closed down in Building 777-M
1980	L	L Reactor restart initiative is launched.
1980	All	SRS begins major program to develop and use metal hydrides in its tritium production facilities.
1980	SRTC (700 Area)	Thermal Cycling Absorption Process (TCAP) is invented at the Savannah River Technology Center.
1981	M	305-M graphite test pile is shut down.
1981	H	Application development begins on the Thermal Cycling Absorption Process (TCAP) to separate hydrogen isotopes.
1981	M	M-Area Settling Basin clean-up project begins.
1982	412-D	Last part of the Heavy Water Rework Facility is permanently closed.
1982	F and H	Work begins to automate canyon processes. Distributed Control System (DCS) is installed in F-Area warm canyon.
1982	All	Tracking Atmospheric Radioactive Contaminants (TRAC) vehicle built.
1983	H	Construction of the Defense Waste Processing Facility begins.
1983	SRTC	Experimental TCAP achieves 97% D_2/H_2 purity.
1984	SRTC	TCAP is selected for application in the new Replacement Tritium Facility (RTF).
1985	HB Line	Production of Pu-238 is initiated.
1985	100-C	C Reactor shuts down for repairs (never restarts.)

Table A-1: Savannah River Site Events Chronology		
Dates	**Areas**	**Activities**
1985	100-L	L Reactor restarts.
1985	H	New HB-Line begins producing Pu-238 for NASA's deep space exploration program.
1985	M	Groundwater remediation system is constructed in M-Area.
1986	Z	Construction of Saltstone begins.
1986	H	Construction of Replacement Tritium Facility using the TCAP process begins.
January 1987	F and H	Construction begins on the F and H Effluent Treatment Project (ETP).
1987	100 Areas	New administrative power level limitations are placed on reactors due to concerns about emergency cooling systems.
1987	F	F-Area A-Line Operations shut down.
August 1988	100-K	K Reactor shuts down for maintenance.
August 1988	100-L	L Reactor shuts down for maintenance and upgrades.
August 1988	100-P	P Reactor shuts down for maintenance.
October 1988	F and H	Effluent Treatment Facility (ETF) begins treatment of wastewater from F- and H-Area separations facilities.
1989	SRTC	TCAP Prototype achieves 99% D_2/H_2 purity.
1989	F	221-F B-Line shuts down.
1990	100-K	Construction of the K Reactor Cooling Tower begins.
1990	Z	Saltstone begins operations.
1991	E	Mixed Waste Management Facility in the Burial Ground Complex is closed.
1991	M	M-Area Settling Basin closure is completed.
December 1991	100-L	L Reactor placed in temporary shuts down status (never restarted.)
July 1992	100-K	K Reactor restarts briefly and is connected to new cooling tower, and is then placed on cold stand-by status.
1992	H	Non-radioactive test operations begin at the Tritium Replacement Facility.
1993	100-K	K Reactor permanently placed in cold-standby condition.
1993	H	Non-radioactive test runs initiated at the Defense Waste Processing Facility.
1993	H	Construction begins on Consolidated Incineration Facility.
1993	F	F Area A-Line restarts.
1993	H	Radioactive operations initiated at the Replacement Tritium Facility.
1994	H	233-H, Replacement Tritium Facility, begins full-scale operations, including unloading gases from reservoirs returned from DOD, separating and purifying the useful tritium and deuterium, mixing the gases to exact specifications, and reloading the reservoirs.
1996	H	Vitrification work begins at the Defense Waste Processing Facility.
1996	100-K	K Reactor is placed on permanent cold shutdown status.
1996	200-F	Nuclear material stabilization work begins with restart of F Canyon facilities.
1997	F	First SRS high-level waste tanks are closed, starting with F-20 Tank.
1997	F	F Area A-Line shuts down permanently.

Table A-1: Savannah River Site Events Chronology		
Dates	**Areas**	**Activities**
2000	100-K	K Reactor Building is converted to K Area Materials Storage Area.
July 2000	H	Groundbreaking for Tritium Extraction Facility.
March 2002	F	Operations end in F Canyon.
2003	F, H Areas	SRS completes the transfer of 30,000 gallons of Am/Cm solutions from F Canyon to H-Area Tank Farm.
February 2005	F	FB Line operations are completed and discontinued.
2005	H	Construction of the Tritium Extraction Facility (TEF) is completed.
October 14, 2005	F	Site preparation begins for the Mixed Oxide (MOX) Fuel Fabrication facility, which will provide for the permanent removal of Pu from potential weapons uses.
November 2006	H	Tritium Extraction Facility begins to extract tritium from rods irradiated in Tennessee Valley Authority (TVA) reactors.
January 2007	H	TEF completes initial extraction of tritium from Tritium Producing Burnable Absorber Rods (TPBARs).
August 1, 2007	F	Construction of the MOX facility begins.

Sources: SRS Highlights, 2000; Bebbington, 1990; DOE/EA-0948; Santos, 2007; Clemons, 1993; DOE, 2008; SRS, 2008; SRS Defense, 2007; SRS F, 2008; SRS FB, 2007; SRS H, 2007; Washington Savannah River Company, 2008; SRS Saltstone, 2007; SRS Spent, 2008; SRS Tritium, 2008; WSRC-SA-11