ORAU Team **Dose Reconstruction Project for NIOSH** Technical Basis Document for the Savannah River Site To Be Used for EEOICPA Dose Reconstructions	Document Number: ORAUT-TKBS-0003 Effective Date: 07/15/2003 Revision No.: 00 Controlled Copy No.: _____ Page 1 of 188
Subject Expert: Edward D. Scalsky Approval: Signature on File Date: 07/15/2003 Judson L. Kenoyer, Task 3 Manager Concurrence: Signature on File Date: 07/15/2003 Richard E. Toohey, Project Director Approval: Signature on File Date: 07/15/2003 James W. Neton, OCAS Health Science Administrator	Supersedes: None

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

RECORD OF ISSUE / REVISIONS

ISSUE AUTHORIIZATION DATE	EFFECTIVE DATE	REV. NO.	DESCRIPTION
Draft	05/10/2003	00-A	Original issue draft technical basis document for the Savannah River Site. Initiated by Judson L. Kenoyer.
Draft	05/19/2003	00-B	Revised to correct formatting errors and make additional text changes. Initiated by Judson L. Kenoyer.
Draft	07/03/2003	00-C	Revised to incorporate changes based on information gathered through contacts with SRS technical subject matter experts. Initiated by Edward D. Scalsky.
Draft	07/14/2003	00-D	Revised to respond to comments provided by NIOSH on July 11, 2003. Initiated by Edward D. Scalsky.
First Approved Issue	07/15/2003	00	Initiated by Edward D. Scalsky.

GLOSSARY

***"A," "B," "C," and "D"*:** codes used in whole body count results; the letters represent energy regions: "A" means 810-910keV, "B" means 920-1020 keV, "C" means 1180-1290 keV, and "D" means 1760-1990 keV.

absorption type: a term introduced in ICRP's revised respiratory tract model (ICRP-66) to replace the clearance classes in ICRP-30. The absorption types are Type F for fast absorption, formerly called Class D; Type M for moderate absorption, formerly called Class W; and Type S for slow absorption, formerly called Class Y. Also referred to as Type.

albedo dosimeter: A dosimeter that measures the slow neutrons which are generated by higher energy neutrons incident on the body and which reflect back into the dosimeter.

annual dose equivalent: the dose equivalent received in a year. The annual dose equivalent is expressed in units of rem (sievert).

Atomic Energy Commission: original agency established for nuclear weapons and power production; a predecessor to the U.S. Department of Energy.

beta (β) dose: a designation (i.e., beta) on some SRS external dose records referring to the dose from less-energetic beta, X ray and/or gamma radiation (see ALSO open window, or shallow dose).

beta radiation: radiation consisting of charged particles of very small mass (i.e., the electron) emitted spontaneously from the nuclei of certain radioactive elements. Most (if not all) of the direct fission products emit beta radiation. Physically, the beta particle is identical to an electron moving at high velocity.

class: The respiratory tract inhalation classification scheme developed in ICRP-30 for inhaled material according to its rate of clearance from the pulmonary region of the respiratory tract. Materials are classified as D (days), W (weeks), or Y (years), according to how fast they clear the lungs; Class D in less than 10 days; Class W in 10 to 100 days; Class Y in more than 100 days. Recent recommendations in ICRP Report 66 replaced classes D, W, and Y with lung absorption Types F (fast), M (moderate), and S (slow).

collective dose equivalent: the sum of the dose equivalents of all individuals in an exposed population. Collective dose is expressed in units of person-rem (person-sievert).

collective effective dose equivalent: the sum of the effective dose equivalents of all individuals in an exposed population. Collective effective dose is expressed in units of person-rem (person-sievert).

curie: a special unit of activity. One curie exactly equals 3.7×10^{10} nuclear transitions per second.

deep absorbed dose (D_d): the absorbed dose at the depth of 1.0 cm in a material of specified geometry and composition.
deep dose equivalent (H_d): the dose equivalent at the respective depth of 1.0 cm in tissue.

detection limit (lower): the minimum quantifiable exposure or neutron flux that can be detected.

delayed neutron analysis (DNA): the method was adopted for both enriched and depleted uranium urinalysis analyses in about 1982.

depleted uranium (DU) analysis: indicates that the chemical properties of uranium were used for analysis, not necessarily that the material in question was depleted uranium

dose equivalent (H): the product of the absorbed dose (D), the quality factor (Q), and any other modifying factors. The special unit is the rem. When D is expressed in Gy, H is in Sieverts (Sv). (1 Sv = 100 rem.)

dose of record: the dose files provided by DOE to NIOSH as part of the individual worker files.

dosimeter: a device used to measure the quantity of radiation received. A holder with radiation-absorbing elements (filters) and an insert with radiation-sensitive elements packaged to provide a record of absorbed dose or dose equivalent received by an individual. (see albedo dosimeter, film dosimeter, neutron film dosimeter, thermoluminescent dosimeter.)

dosimetry: the science of assessing absorbed dose, dose equivalent, effective dose equivalent, etc., from external and/or internal sources of radiation.

dosimetry system: a system used to assess dose equivalent from external radiation to the whole body, skin, and/or extremities. This includes the fabrication, assignment, and processing of the dosimeters as well as interpretation and documentation of the results.

DU*:* as used in this document depleted uranium, used as targets for ^{239}Pu production; isotopic activity ratios (NOTE: this is not the mass ratio) listed by SRS as:

Isotope	Activity fraction
^{234}U	0.0840
^{235}U	0.0145
^{238}U	0.9015

enriched uranium (EU) analysis: as used in this document indicates the analysis was by alpha counting, not necessarily that the material in question was enriched uranium

exchange period (frequency): time period (weekly, biweekly, monthly, quarterly, etc.) for routine exchange of dosimeters.

exposure: as used in the technical sense, exposure refers to a measure expressed in roentgens (R) of the ionization produced by photon radiation (i.e., gamma and X rays) in air.

extremity: that portion of the arm extending from and including the elbow through the fingertips, and that portion of the leg extending from and including the knee and patella through the tips of the toes.

FB-Line: the plutonium finishing line located in the 221-F facility.

Field (f): refers to SRS practice to quantify Neutron Track Emulsion Type A (NTA) neutron tracks using a projection image covering a 15-inch projection screen.

field calibration: dosimeter calibration based on radiation types, intensity and energies present in the work environment.

film: generally means a "film packet" that contains one or more pieces of film in a light-tight wrapping. The film when developed has an image caused by radiation that can be measured using an optical densitometer. (see Dupont 552, Dupont 558, Eastman Kodak, Nuclear Emulsions.)

film density: see optical density.

film dosimeter: a small packet of film within a holder that attaches to a wearer.

flux density (n/cm^2-sec): a measure of the intensity of neutron radiation in neutrons/cm^2-sec. It is the number of neutrons passing through 1 square centimeter of a given target in 1 second. Expressed as nv, where n = the number of neutrons per cubic centimeter and v = their velocity in centimeters per second.

gamma rays (G or γ): electromagnetic radiation (photons) originating in atomic nuclei and accompanying many nuclear reactions (e.g., fission, radioactive decay, and neutron capture). Physically, gamma rays are identical to X rays of high energy, the only essential difference being that X rays do not originate in the nucleus.

Gray (Gy): The special name for the SI unit of absorbed dose. (1 Gy = 1 Jkg^{-1})

HB-Line: the plutonium finishing line located in the 221-H facility.

HEU: highly enriched uranium, used in Naval Reactor Fuel; isotopic ratios listed by SRS as:

Isotope	Activity fraction
^{234}U	0.9806
^{235}U	0.0194
^{238}U	0.0000

Health Protection Radiation Exposure Database (HPRED): this database replaced the mainframe tape master file system in April 1989. This system interfaced with the TLD badge reader system. In 1990, visitor records were included in this database.

Health Protection Annual Radiation Exposure History Database (HPAREH): Starting in 1980, this database was used to generate yearly radiation exposure reports to employees.

ionizing radiation: electromagnetic or particulate radiation capable of producing charged particles through interactions with matter.

liquid scintillation counter (LSC): used for tritium urinalysis starting in 1958.

maximum permissible body burden (MPBB): limit for body content of a radionuclide used from startup until 1975

maximum permissible lung burden (MPLB): during the 1970's the occupational limit for Pu was expressed in terms of a quantity of plutonium that could be present in the chest at any given time, equal to 16 μCi (0.25 μg) ^{239}Pu.

maximum permissible organ burden (MPOB): limit for organ content of a radionuclide used from startup until 1975.

neutron: a basic particle that is electrically neutral, having nearly the same mass as the hydrogen atom.

neutron, fast: neutrons with energy equal or greater than 10 keV.

neutron, intermediate: neutrons with energy between 0.5 eV and 10 keV.

neutron, thermal: strictly, neutrons in thermal equilibrium with surroundings. Generally, neutrons with energy less than about 0.5 eV.

neutron film dosimeter: a film dosimeter that contains an Neutron Track Emulsion, type A, film packet.

nuclear emulsion: often referred to as “NTA” film and used to measure personnel dose from neutron radiation.

Nuclear Track Emulsion, Type A (NTA): a film that is sensitive to fast neutrons. The developed image has tracks caused by neutrons that can be seen by using an appropriate imaging capability such as oil immersion and a 1000X power microscope or a projection capability.

open window (OW): designation on SRS film dosimeter reports that implies the use of little (i.e., only security credential) shielding. It commonly is used to label the film response corresponding to the open window area.

operating area: designation of SRS major operational work areas among the respective fuel fabrication (300), reactor operations (R, P, L, K and C), chemical separations (F and H), uranium separation (A-Line), plutonium finishing (H-Canyon, B-Line), research and development, and transportation, communication and general site support.

optical density: the quantitative measurement of photographic blackening; the optical density is defined as $D = \mathrm{Log}_{10} (I_o/I)$.

Parameter 1: the column in the IREP template where the dose reconstructor will enter the calculated dose. Multiple entries based on year of employment, type of radiation and appropriate energy ranges, internal and external exposure are possible.

Parameter 2: the column in the IREP template where the dose reconstructor will enter the lower limit of the dose distribution based on the radiation type and the dose distribution type.

PuF_4 source: a neutron source whose activating material is plutonium fluoride. The source was used to duplicate the neutron energies in SRS plutonium handling facilities.

pencil dosimeters: a type of ionization chamber used by personnel to measure radiation dose. These results may be labeled as “Pen” dose. Other names: pencil, pocket dosimeter, pocket pencil, pocket ionization chamber (PIC).

personal dose equivalent $H_p(d)$: represents the dose equivalent in soft tissue below a specified point on the body at an appropriate depth d. The depths selected for personnel dosimetry are 0.07 mm and 10 mm, respectively, for the skin and body. These are noted as $H_p(0.07)$ and $H_p(10)$, respectively.

photon: a unit or "particle" of electromagnetic radiation consisting of x- and/or gamma rays.

photon - X ray: electromagnetic radiation of energies between 10 keV and 100 keV whose source can be an X-ray machine or radionuclide.

quality factor, Q: a modifying factor used to derive dose equivalent from absorbed dose.

Rad: the unit of absorbed dose

radiation: alpha, beta, neutron, and photon radiation.

radioactivity: the spontaneous emission of radiation, generally alpha or beta particles, gamma rays, and neutrons from unstable nuclei.

Radiological Qualification Badge (RQB): System starting in 1992 to control each worker’s dosimetry requirements including bioassay requirements. Each radiation worker carries his/her personal card. From 1992 to 1999 the card listed requirements that had been met by the worker prospectively prior to entering radiation areas. That is, the card requirements and the Radiation Work Permit requirements had to match prior to entry. From 1999 to present the requirements on the card are listed retrospectively based on entries made during the designated period. Prior to the RQB system, an electronic database was used to track worker dosimetry requirements.

relative biological effectiveness (RBE): a ratio of the absorbed dose of a reference radiation to the absorbed dose of a test radiation producing the same biological effects, other conditions being equal.

rem: the rem is a unit of dose equivalent, which is equal to the product of the number of rad absorbed and the "quality factor."

rep: a historical unit, roentgen-equivalent-physical (mrep = millirep) used when reporting beta exposures, usually recorded in mrep; equivalent to 93 ergs of energy per gram.

Roentgen (R or r): a unit of exposure to gamma (or X ray) radiation. It is defined precisely as the quantity of gamma (or x) rays that will produce a total charge of 2.58×10^{-4} coulomb in 1 kg of dry air STP. An exposure of 1 R is approximately equivalent to an absorbed dose of 1 rad in soft tissue for higher (~>100 keV) energy photons.

RU: recycled uranium, is SRS reactor fuel; isotopic activity ratios listed by SRS as:

Isotope	Activity fraction
^{234}U	0.8489
^{235}U	0.0120
^{236}U	0.1388
^{238}U	0.0003 or 0.0004 (both listed)

shallow absorbed dose (D_s): the absorbed dose at a depth of 0.07 mm in a material of specified geometry and composition.

shallow dose equivalent (H_s): dose equivalent at a depth of 0.07 mm in tissue.

shielding: any material or obstruction that absorbs (or attenuates) radiation and thus tends to protect personnel or materials from radiation.

shield: designation on SRS dose records that imply a radiation dose, typically to the whole body, from higher energy photon radiation.

sievert (Sv): the SI unit for dose equivalent. (1 Sv = 100 rem.)

silver shield(s): the 1-mm thick shields covering the film packet in the early SRS personnel film dosimeters.

skin dose: absorbed dose at a tissue depth of 7 mg/cm^2 ~ 0.07 mm is tissue..

Snoopy: portable neutron monitoring instrument with a moderated BF_3 detector.

SP: this historical designation stands for “Special Product” and referred to ^{237}Np.

thermoluminescence: property of a material that causes it to emit light as a result of being excited by heat.

thermoluminescent dosimeter (TLD): a holder containing solid chips of material that when heated will release the stored energy as light. The measurement of this light provides a measurement of absorbed dose.

3% Plutonium, 6% Plutonium, and 12% Plutonium: designations for typical mixtures of plutonium encountered at SRS, where the percentage represents the percent by mass of ^{240}Pu in the mixture.

whole body dose: commonly defined as the absorbed dose at a tissue depth of 1.0 cm (1000 mg/cm^2); however, this term is also used to refer to the recorded dose.

X ray: ionizing electromagnetic radiation of external nuclear origin or a radiograph

ACRONYMS AND ABBREVIATIONS

α	alpha particle
AEC	(U.S.) Atomic Energy Commission
AED	Atomic Energy Division (of SRP)
AECL	Administrative Exposure Control Level
AEDE	Annual Effective Dose Equivalent
ARU	Acid Recovery Unit
AWE	Atomic Weapons Employer
β	beta particle
BSRI	Bechtel Savannah River Incorporated
BSRI	Bechtel Savannah River Institute
BTM	Berthold Tritium Monitor
CATI	Computer Assisted Telephone Interview
CEDE	Committed Effective Dose Equivalent (replaced the AEDE in 1992)
c/m	counts per minute
CND	Criticality Neutron Dosimeter
DAC	Derived Air Concentration
DAC-h	Derived Air Concentration hours
DCF	Dose Conversion Factor
DNA	Delayed Neutron Analysis
DE	Dose Equivalent
DL	Detection Limit
DOE	U. S. Department of Energy
DOELAP	DOE Laboratory Accreditation Program
DOL	Department of Labor
DPSOP	Du Pont Standard Operating Procedure
DWPF	Defense Waste Processing Facility
DU	Depleted Uranium
EEOICPA	Energy Employees Occupational Illness Compensation Program Act
ESE	Entrance Skin Exposure
ERDA	Energy Research & Development Administration
ETF	Effluent Treatment Facility
γ	gamma ray
HEU	Highly Enriched Uranium
HHS	Department of Health and Human Services
HHE	High Heat Waste
HLW	High Level Waste
H_p (0.07)	Dose equivalent at 0.07 mm depth
$H_p(10)$	Personal Dose Equivalent at 10 mm depth
HPRED	Health Protection Radiation Exposure Database
HPAREH	Health Protection Annual Radiation Exposure History Database
HRTM	Human Respiratory Tract Model

HTO	Tritiated Water
HVL	Half Value Layer
IARC	International Agency for Research on Cancer
ICRP	International Commission on Radiological Protection
IMBA	Integrated Modules for Bioassay Analysis
IREP	Interactive RadioEpidemiological Program
KPA	Kinetic Phosphorimetry Analysis
LANL	Los Alamos National Laboratory
LLD	Lower Limit of Detection
LSC	Liquid Scintillation Counter
LHW	Low Heat Waste
MDA	Minimum Detectable Activity
MDL	Minimum Detection Limit
MED	Manhattan Engineering District
MEPAS	Multimedia Environmental Pollutant Assessment System
MPBB	Maximum Permissible Body Burden
MPLB	Maximum Permissible Lung Burden
MPOB	Maximum Permissible Organ Burden
NCRP	National Council on Radiation Protection and Measurements
NIOSH	National Institute for Occupational Safety and Health
NOCTS	NIOSH OCAS Claims Tracking System
NTA	Nuclear Track Emulsion
OCAS	(NIOSH) Office of Compensation Analysis and Support
ORAU	Oak Ridge Associated Universities
ORNL	Oak Ridge National Laboratory
PA	Posterior-Anterior (medical X ray)
PAS	Personal Air Sampler
PDIS	Personnel Dosimetry Intercomparison Studies
PEF	$^{238}PuO_2$ Experimental Facility
PIC	Pocket Ionization Chamber (i.e., “Pencil” dosimeter)
PSF	Plutonium Storage Facility
PuFF	$^{238}PuO_2$ Fuel Form Facility
PUREX	Plutonium Uranium Extraction (Facility/Process)
R	Roentgen
RAS	Regional Air Sampler
RBE	Relative Biological Effectiveness
RBOF	Receiving Basin for Offsite Fuel
RC & HP	Radiological Control and Health Physics Department
rem	Roentgen equivalent man
rep	Roentgen equivalent physical
RMS	Root Mean Square

RQB:	Radiological Qualification Badge
RRF	Resin Regeneration Facility
RTF	Replacement Tritium Facility
RU	Recycled Uranium
SEC	Special Exposure Cohort
SSD	Source to Skin Distance
SID	Source to Image Distance
SMI	Stack Monitor Integrator
SRI	Savannah River Institute
SRL	Savannah River Laboratory
SRO	Savannah River Operations Office
SRP	Savannah River Plant
SRS	Savannah River Site
STM	Stack Tritium Monitor
SWDF	Solid Waste Disposal Facility
TEPC	Tissue Equivalent Proportional Counter
TBD	Technical Basis Document
TLD	Thermoluminescent Dosimeter
TLND	Thermoluminescent Neutron Dosimeter
TPO	Technical Procedures Office
TRU	Transuranic
WCF	Waste Certification Facility
WSRC	Westinghouse Savannah River Company

EXECUTIVE SUMMARY

In enacting the Energy Employees Occupational Illness Compensation Program Act of 2000 (EEOICPA), the U.S. Congress officially recognized the hazardous nature of producing and testing nuclear weapons. Under the Act workers, who have developed selected types of cancer or their survivors may be entitled to compensation and medical benefits. This program is administered by the Department of Labor (DOL) Office of Worker Compensation (OWCP). The Department of Health and Human Service's (HHS) National Institute for Occupational Safety and Health (NIOSH) is responsible for determining the individual worker's dose.

Specifically the Act requires the estimation of radiological doses from ionizing radiation received by workers in the nuclear weapons production programs of the various U. S. Department of Energy (DOE) and its predecessor agencies. Methods for implementing provisions of the Act have been promulgated in 42 Code of Federal Regulations Part 82 (40CFR Part 82), "Methods for Radiation Dose Reconstruction Under the Energy Employees Occupational Illness Compensation Program Act of 2000" (Federal Register, Vol. 67 No. 85, Thursday, May 2, 2002).

Oak Ridge Associated Universities (ORAU) leads a team, the ORAU Team, to support NIOSH in conducting this major program. This technical basis document (TBD) represents a specific support mechanism to the ORAU Team concerning documentation of historical practices at the Savannah River Site (SRS). This TBD can be used to evaluate both internal and external dosimetry data for unmonitored and monitored workers and serve as a supplement to, or substitute for, individual monitoring data. This document provides a Savannah River Site (SRS) profile that contains technical basis information used by the ORAU Team to evaluate the total occupational radiation dose for Energy Employees Occupational Illness Compensation Program Act (EEOICPA) claimants.

This document also provides supporting technical data to evaluate, with claimant-favorable assumptions, the total SRS occupational radiation dose that may reasonably be associated with the worker's radiation exposure. This dose results from exposure to external and internal radiation in SRS facilities; to SRS occupationally-required diagnostic x-ray examinations, and to on-site environmental releases. Also included is the dose that may have occurred while the worker was unmonitored or the dose may have been missed. Over the years new and more reliable scientific methods and protection measures have been developed. The methods needed to account for these changes are also identified in this document.

The doses are evaluated using the NIOSH Interactive RadioEpidemiological Program (IREP) and the Internal Modular Bioassay Analysis (IMBA) computer codes. Information on measurement uncertainties is an integral component of the NIOSH approach. This document describes how the uncertainty for SRS exposure and dose records is evaluated.

The document is divided into five major sections; Facilities and Processes, Occupational Medical Dose, Occupational Environmental Dose, Occupational Internal Dose, and Occupational External Dosimetry. Each section has an associated attachment, which provides the critical data for the specialists reconstructing the doses.

The facilities and processes used at SRS over the years and their associated source terms are briefly described in section one. The tables showing the radionuclides of concern for dose reconstruction and the years of operation of the different facilities are in Attachment A. The discussion includes the individual processes and the radionuclides for both the internal and external exposure potentials. These data would be used by the dose reconstructors only when monitoring data are unavailable and/or other methods may not be amenable for claimant dose reconstruction.

Section 2 discusses the occupational medical dose. The medical doses result from the chest X rays given during pre-employment and annual physical exams. However, certain employees may have had X rays more frequently based on their work and their age. Although X rays may have been given because of on-the-job injuries, these doses are not considered part of the occupational dose. The x-ray techniques and equipment used over the years have also changed and are presented and discussed in this section. Attachment B presents the pertinent tables for use by the dose reconstructors in determining the worker's occupational medical doses.

Section 3 provides information for calculating internal and external radiation doses to unmonitored workers at the SRS. Data are presented on the radionuclides released from the site facilities and processes. The methodology to calculate the doses received by unmonitored workers outside the facilities is described. Internal doses result from breathing the radioactive material in the air, ground level releases, or resuspended from radioactive material deposited on the ground. External doses result from the ambient radiation levels and from submersion in a plume of noble gases. The equipment used to measure both the internal and external doses changed considerably over the years and these changes have been taken into account in calculating the individual worker dose.

A dispersion model was developed using a 5-year average joint frequency distribution. The values were converted to X/Q tables for elevated and ground level releases, using the Multimedia Environmental Pollutant Assessment System (MEPAS) 4.0 computer code, and plotted using an Excel spreadsheet. Various assumptions were made to ensure atmospheric dispersion values were not being overestimated. The maximum air concentrations were then calculated for the principal dose producing radionuclides at various on-site populated areas for use in calculating the unmonitored worker doses.

Attachment C provides a series of tables of the 50^{th} percentile and geometric standard deviation (GSD) intake values for selected individual radionuclides resulting from releases from various facilities. These values will be used by the dose reconstructors to calculate the internal and external doses that unmonitored workers may receive from the environmental pathway.

Section 4 discusses the occupational internal dose. The radionuclides identified by Savannah River as being the most significant in delivering doses or that were effective tracers for significant dose-producing radionuclides are presented. The methods for analysis of selected individual radionuclides are also discussed. Tables providing the minimum detectable activities (MDAs) are included for the *In vitro* analysis of urine and feces for selected radionuclides and *in vivo* whole body and chest counting.

Attachment D provides a more complete list of radionuclides and MDAs for use by dose reconstructors for calculating internal doses.

Section 5 presents the occupational external dosimetry program for measuring skin and whole body doses to the workers. The methods for evaluating external doses to workers have also evolved over the years as new techniques and equipment have been developed. Concepts in radiation protection have also changed. The dose reconstruction parameters, SRS practices and policies, and dosimeter types and technology for measuring the dose from the different types of radiation are discussed in this section. Attention is given to the evaluation of doses measured from exposure to beta, gamma, and neutron radiation. Test results for various films exposed to different geometries and energies of radiation are given in tables.

Sources of bias, workplace radiation field characteristics, responses of the different beta/gamma and neutron dosimeters in the workplace fields, and the adjustments to the recorded dose measured by these dosimeters during specific years are discussed in detail.

There are sources of potential dose that could be missed because of the limitations of dosimetry systems and the methods of reporting low doses. This missed dose is discussed as a function of facility location, dosimeter type, year, and energy range. Attachment E describes the use of the external dosimetry technical basis parameters to facilitate the efforts of the dose reconstructors.

1.0 INTRODUCTION

The National Institute for Occupational Safety and Health (NIOSH) was assigned the responsibility for developing the technical capabilities and guidance to be used to implement the Energy Employees Occupational Illness Compensation Program Act (EEOICPA). Oak Ridge Associated Universities (ORAU) led a team, identified as the ORAU Team, to support NIOSH in the conduct of this major program. This technical basis document (TBD) represents a specific area of support to the ORAU Team concerning documentation of historical practices at the Savannah River Site (SRS). It pertains to evaluation of both internal and external dosimetry data for unmonitored and monitored workers and is to be used as a supplement to or substitute for individual monitoring data.

1.1 Purpose

The purpose of this document is to provide a Savannah River Site (SRS) profile that contains technical basis information used by the Oak Ridge Associated Universities (ORAU) Team to evaluate the total occupational dose for EEOICPA claimants.

1.2 Scope

SRS operations played an important role in the US nuclear weapons program (DOE 1997). These SRS processes included nuclear fuel fabrication, reactor operation, radiochemical processing, uranium recycling, plutonium production, neutron source production, and waste management. Supporting documentation has been assembled in this TBD to assist in the evaluation of worker dose from these processes using the methodology in the respective NIOSH OCAS-IG-001 External Dose Reconstruction Implementation Guideline (NIOSH 2002a) and OCAS-IG-002 Internal Dose Reconstruction Implementation Guideline (NIOSH 2002b).

The methods and concepts of measuring radiation exposure to workers have evolved since the beginning of SRS operations in 1952. An objective of this document is to provide supporting technical data to evaluate, with claimant-favorable assumptions, the total SRS occupational dose that may reasonably be associated with worker radiation exposure as covered under EEOICPA legislation. This dose includes occupational external and internal exposure in SRS facilities; SRS occupationally-required diagnostic X-ray examinations, and on-site exposure to SRS environmental releases. The documentation addresses evaluation of unmonitored and monitored worker exposure, and missed dose. Consistent with NIOSH Implementation Guidelines, this document identifies how to adjust the historical occupational dose to account for current scientific methods and protection factors.

This document also presents the technical basis of methods used to prepare the SRS worker dose records for input to the NIOSH Interactive RadioEpidemiological Program (IREP) and the Integrated Modules for Bioassay Analysis (IMBA) computer codes used to evaluate worker dose. Because information on measurement uncertainties is an integral component of the NIOSH approach, this document describes how the uncertainty for SRS exposure and dose records is evaluated.

The main body of text in this document provides the description of the facilities and processes, historical information related to worker internal and external exposures, and environmental data for use when actual monitoring data are unavailable. The attachments represent the critical data and tables required by dose reconstructors for performing the individual claimant dose reconstructions. These attachments should suffice as a stand-alone document for dose reconstruction. Additional details, if necessary, could be found in the main body of the text.

1.3 Site Activities and Processes

This section briefly describes the facilities and processes while their associated source terms have been captured in Attachment A. These data would be used by dose reconstructors only when monitoring data are unavailable and/or other methods may not be amenable for claimant dose reconstruction. As indicated, the tables listed below and referenced in the text refer to the tables in Attachment A.

The following tables or sections are provided in Attachment A:

Table Number	Facility or Process	Page Number
A-1.1	C, L, K, P and R Reactors-Years of Operation	
A-1.2	Radionuclides of Concern for All Reactors	
A-2	F-Area A-Line Facility	
A-3	221-F B-Line Facility	
A.4-1	221-F Canyon Facility, Head End Stream (Stream 1)	
A-4.2	221-F Canyon Facility, (2^{nd} Uranium Cycle) (Stream 2)	
A-4.3	221-F Canyon Facility, (2^{nd} Cycle Plutonium) (Stream 3)	
A-5	New Special Recovery (NSR) Facility (Pu Recovery)	
A-6.1	F-Area Outside Facilities, Acid Recovery Unit (ARU)	
A-6.2	F-Area Outside Facilities, Water Handling System	
A-6.3	F-Area Outside Facilities, General Purpose Evaporators	
A-6.4	F-Area Outside Facilities, Segregated Plutonium Solvent	
A-6.5	F-Area Outside Facilities, Laboratory Waste	
A-6.6	F-Area Outside Facilities, Segregated Uranium Solvent	
A-7	$^{238}PuO_2$ Fuel Form Facility (PuFF) and $^{238}PuO_2$ Experimental Facility (PEF)	
A-8	235-F Vaults (Pu Oxide, Finished Product, Scrap Vaults)	
A-9	772-F and 772-1F Production Control Laboratories	
A-10	E-Area Solid Waste Disposal Facility (SWDF)	
A-11	Plutonium Storage Facility (PSF)	
A-12	F/H Effluent Treatment Facility (ETF)	
A-13	F & H Area Cooling Water and Retention Basins	
A-14	F & H Area Tank Farms	
A-15	Waste Certification Facility (WCF)	
A-16.1	221-H B-Line Facility, Scrap Recovery U-238, U-239	
A-16.2	221-H B-Line Facility, Scrap Recovery U-235	
A-16.3	221-H B-Line Facility, NpO_2 and PuO_2 Facility	
A-17.1	H-Canyon Facility, First Cycle Stream-Uranium Cycle	
A-17.2	H-Canyon Facility, Second Uranium Cycle Stream	
A-17.3	H-Canyon Facility, Neptunium Second Cycle	
A-17.4	H-Canyon Facility, Work Stream	
A-18.1	H-Area Outside Facilities, A Line	
A-18.2	H-Area Outside Facilities, Water Handling System	
A-19.1	Receiving Basin for Offsite Fuel (RBOF)	
A-19.2	Resin Regeneration Facility (RRF)	
A-20	Tritium Facilities	
A-21	Uranium Target Fabrication Facility, 313 M	
A-22	Uranium Processing Facilities 320-M, 322-M, and 341-M	
A-23	Fuel Fabrication Facility, 321-M	
A-24.1	S-Area Defense Waste Processing Facility (DWPF), Sludge Slurry Feed Stream (Stream 1)	
A-24.2	S-Area Defense Waste Processing Facility (DWPF), Precipitate Slurry Feed Stream (Stream 201)	
A-25.1	Z Area, Waste Tank Supernate - Decontaminated Salt Solution	
A-25.2	Z Area, Saltstone	
A-26	A Area Building 773-A	
A-27	Buildings 735-A and 735-11A	
A-28	Building 776-A Liquid Waste Handling Facility	
A-29	D-Area, Heavy Water Production and Reprocessing	

The data in the tables in Attachment A have all been extracted from ESH-HPT-960197, Facility Descriptions (U), Thomas R. La Bone, October 1, 1996.

All stated assumptions are also from La Bone (1996).

The following definitions apply to all tables in Attachment A that contain information on the Activity Fraction:

The **activity fraction** is the fraction of the total activity represented by a particular radionuclide.

The units in the tables in Attachment A are conventional units. To convert from mrem/nCi to Sv/Bq, multiply by 2.7E-7.

To convert from nCi to Bq, multiply by 37.

Where solubility classes were available in the Savannah River Site literature, their designations have been changed to the ICRP absorption type designations in the tables. According to the International Commission on Radiological Protection ICRP Publication 68 “Dose Coefficients for Intakes of Radionuclides by Workers” the ICRP 30 D, W, and Y absorption classes correspond broadly to the absorption Types F, M, S respectively. These values are given in the second column of the tables in Attachment A.

The differences in the radionuclides for the various processes in Attachment A are related to the mode of exposure. To minimize the number of pages that the dose reconstructor would need to search, both internal and external radionuclides have been combined into one table for each of the facilities/processes addressed for the SRS. Radionuclides identified as “being of concern” for internal exposure may be different from those identified as providing the greatest external exposure. The last column of some of the tables is “Significant to External Exposure.” An “X” in this column indicates that the radionuclide contributes to external dose. While ^{90}Sr is typically an internal exposure hazard, under certain circumstances it could also be an external exposure hazard. It is listed in this document as both for completeness.

Dates of operation of the various facilities have been taken from:

Cummins, et al, Radioactive Releases from the Savannah River Site, 1954 – 1989, WSRC-RP-91-684,

DOE/EM-0319, Linking Legacies, January, 1997, and

Till, et al, Savannah River Site Environmental Dose Reconstruction Project, Phase II, Risk Assessment Corporation, 30 April 2001.

The discussions that follow relate directly to the table represented in the individual section heading, e.g., A-1 Reactors will relate to Table A-1 in Attachment A, etc.

A-1 Reactors (LaBone 1996, Till 1999)

Five heavy-water reactors designated R, P, L, K, and C were constructed at SRS in the early 1950s, each of which is in the locations designated R-Area, P-Area, L-Area, K-Area, and C-Area respectively. All five reactors were operating in 1955. Reactor shutdown began in 1964 with R-Reactor. C-Reactor was shut down in 1987, L-Reactor was shut down in 1968, restarted in 1985, and then shut down again in 1988. K-Reactor and P-Reactor operated with few interruptions until 1988. A restart of K-Reactor began in 1991, but was not completed. None of the reactors are currently operational.

Depleted uranium, ^{237}Np, and lithium targets were irradiated to produce ^{239}Pu, ^{238}Pu, and tritium, respectively. The targets are irradiated and transferred to the Tritium Facilities, H Area Separations (^{238}Pu), and F Area Separations (^{239}Pu) for processing. Spent reactor fuel is sent to H Area Separations for processing. In the past the reactors were also used to convert thorium to ^{233}U as well as to irradiate targets that produced transplutonic elements such as curium and californium.

The composition and relative abundance of the radionuclides in the reactor environment are not known. Representative samples are taken annually to determine the isotopic makeup of contamination in reactor areas. When the reactors are operating, fission neutrons are an external exposure concern in some areas. For the purpose of determining internal or external exposure in the reactor areas, the radionuclides of concern in the reactor areas are presented in Table A.1-1.

Internal exposure - Discussions by T.R. La Bone with Health Physics Operations personnel and a review of ANSI 343 were used to identify the radionuclides of concern. Tritium is assumed to produce most of the personnel exposure because of the large quantities present.
Because the reactors were heavy-water reactors, intakes from the reactor buildings would be so dominated by tritium that intakes of fission/activation products can be ignored. If, in a specific case, the bioassay data indicate otherwise, then see the fission product discussion for the 221-F Canyon Facility.

External exposure - During operation, fission neutrons are also a concern in certain areas.

A-2 F-Area and A-Line Facility

The A-Line facility accepts dilute, depleted uranyl nitrate solution from the 221-F Facility and converts it into uranium trioxide powder. Dilute uranyl nitrate solution is sent to A-Line from the solvent extraction process in the 221-F Facility where it is initially decanted to remove tributyl phosphate n-paraffin solvent. It is then concentrated by evaporation. A second evaporation yields a hydrate of uranyl nitrate that is heated in a denitrator to yield the uranium trioxide product.

The A-Line facility also has a dissolver to prepare a uranyl nitrate solution from the uranium trioxide product and nitric acid when it is required as a process feed stream back into the solvent extraction process in the 221-F Facility. The dissolver is also used to recycle off-standard A-Line product. The uranium trioxide product is placed into drums for storage on-site or into packages for off-site shipment.

The A-Line Facility can be broken down into unit operations that perform specific functions in the creation of the final product. These processes consist of:

- evaporation, that accepts uranyl nitrate from the 221-F Warm Canyon and concentrates it;
- purification, which is an optional step, but by using silica gel absorbs residual zirconium and niobium out of the uranyl nitrate solution;
- hydrate evaporation, where uranyl nitrate from the silica gel or evaporator is again concentrated;
- denitration, where material from the hydrate evaporation is received and converted from uranyl nitrate hexahydrate to uranium trioxide;
- material handling, where the uranium trioxide powder is removed from the denitrators and placed into storage bins from which it is loaded into containers, and
- oxide dissolution, where 221-F occasionally requires a uranyl nitrate-nitric acid solution for use as a process stream.
- The fume recovery system converts oxides of nitrogen into nitric acid for reuse using exhausters, a venturi scrubber, off-gas coolers, an absorption column, and various pumps and tanks. (The venturi scrubber removes uranium trioxide particles from the denitrator off-gas and is the only piece of the system that is of radiological concern.)

Internal exposure - The chemical forms that are produced in A-Line are uranyl nitrate and uranium trioxide. These materials are Types F and M, respectively. See also the discussion in 4.1.2, "In Vitro Bioassay for Uranium," for guidance on interpreting uranium urinalyses for natural or depleted uranium. Although the trace contaminants in the uranium provide little to "compliance" dose (i.e., 50-year committed doses), they may contribute dose to different organs than the uranium and they will be more significant if the time from intake to diagnosis is short relative to 50 years. Hence, the trace radionuclides should be included in intake calculations in the ratio relative to ^{238}U as shown in Table A.2.

External exposure - Depending on the age of the material in the A-Line, it is assumed a mixture of long-lived fission products would be the major contributors of external dose. The long-lived fission products of interest would include ^{95}Nb, ^{103}Ru, ^{106}Ru, ^{141}Ce, ^{144}Ce, ^{144}Pr, ^{151}Sm, and ^{155}Eu. Other contributors would be ^{235}U, ^{236}U, ^{238}U, and ^{239}Pu.

The activity fractions given in the table indicate the predominant radionuclides of concern for external dose would be ^{238}U, ^{141}Ce, ^{144}Ce, ^{103}Ru, ^{106}Ru, and ^{95}Nb.

A-3 221-F B-Line Facility

The F B-Line Facility is located in the 200-F Separations Area and is used to convert plutonium nitrate into plutonium metal or plutonium oxide. It also has the capability to recover plutonium from on-site and off-site scrap. Initially the plutonium produced in this facility is primarily ^{239}Pu in a dilute nitric acid solution. From there it undergoes many chemical processes that produce an end product.

The individual processes in the 221-F B-Line Facility consist of:

- Cation exchange, where dilute plutonium solution transferred from the canyon storage tanks to the F B-Line receiving tanks is sent through a cation exchange column that removes the plutonium along with some impurities;

- Drying and conversion, where the plutonium trifluoride taken from the filtration process is dried and roasted to produce plutonium dioxide and plutonium tetrafluoride;
- Reduction, where the plutonium dioxide-plutonium tetrafluoride mixture is weighed, mixed with metallic calcium and placed into a magnesium crucible whose inside is filled with magnesium oxide sand, sealed, pressurized with argon, and heated to a molten state, Pu metal flows to the bottom of the crucible and forms a button shaped solid, which is then cooled, separated from the calcium slag, and die-stamped for identification;
- Button finishing, where the stamped button is submerged in nitric acid, rinsed in water and dried (after which one or two grams of the plutonium button are removed using a drill), placed in a can, crimp-sealed, placed in a plastic bag, a second can, crimp-sealed, and sent to storage.

The drill turnings are sent to the 772-F Laboratory for analysis and recovery, where plutonium is extracted from the solid scrap from on-site and off-site sources, as well as miscellaneous solutions from F B-Line by using an anion exchange resin. The scrap is initially dissolved using a mixture of aluminum nitrate and nitric acid, after which the plutonium scrap is sorbed on the anionic resin and sent back to the canyon for processing.

External exposure considerations - The alpha emitting isotopes of plutonium are not usually an external exposure concern. However, when they exist in a fluoride form, the alpha particles interact with the fluorine by an (α, n) reaction to produce a significant source of neutrons. Therefore, neutrons are a major external exposure concern in the F B-Line.

Neutrons are also emitted by spontaneous fission along with associated beta and gamma radiations from ^{238}Pu and ^{240}Pu and progeny. Plutonium-241 decays by beta emissions to ^{241}Am which emits a 60 keV photon and X rays with an energy of 17 keV. These photons are the primary external exposure concerns for ^{241}Pu.

Many different mixtures of plutonium isotopes are possible in the 221-F B-Line, but it is assumed that the mixture will be either 6% or 12% Pu. The percent refers to the mass percent of ^{240}Pu. The radionuclides of concern in Table A-3 are the same for both internal and external exposure.

Internal exposure - the plutonium chemical compounds in the process are varied, and the lung inhalation/solubility types vary with these compounds so Table A-3 does not list a specific type. If the bioassay data are not robust enough to imply the type, the dose reconstructor should assume type M for nitrate form and type S for oxide form. However, old contamination, even if originally in the nitrate form, slowly oxides and could be considered type S. If no information is available concerning the chemical form, assume a type that is claimant-favorable; type M will increase the dose to systemic organs; type S will increase the dose to the lung and GI track organs.

A-4 221-F Canyon facility

The 221-F Canyon Facility consists of two parallel canyons that constitute the process area. These canyons contain the high-activity and low-activity radioactive materials and are referred to as the hot and warm canyons, respectively.

The F-Canyon facility uses the Plutonium Uranium Extraction (PUREX) process to recover ^{239}Pu, ^{237}Np, and ^{238}U from reactor-irradiated targets. The plutonium is transferred to the B-Line and the uranium is transferred to the A-Line facility where they are processed into a solid form.

The 221-F Canyon processes are:

Head End
: Irradiated targets are received from the reactor areas. The aluminum cladding is dissolved in hydroxide-sodium nitrate solution, and the uranium metal is then dissolved in nitric acid.

Solvent Extraction
: The first solvent extraction process separates the plutonium and the uranium from the fission products using tributyl phosphate in a hydrocarbon diluant. In the second cycle, plutonium and uranium are separated and purified.

Anion Exchange
: The aqueous waste produced in the first solvent extraction cycle and the second uranium cycle contain neptunium and plutonium in a nitric acid solution, which are recovered using an anion exchange resin column. After the solution is sent through the column, the diluted actinides are sent to the first solvent extraction cycle or warm canyon for further processing and separation. If the concentration of the actinides is such that the solution is sent to the warm canyon, a further series of columns are used to purify and separate the plutonium from the neptunium. At this stage, ^{232}Th and ^{234}Th are also removed.

Waste Evaporators
: Aqueous waste from the first extraction cycle, the second uranium cycle, acid strip condensate, ventilation system sumps, and concentrate from the second stage high activity waste evaporator are all combined and evaporated in the first stage continuous evaporator. This process reduces the volumes of waste going to the high level waste storage tanks.

Locations of Increased Exposure Potential
: The hot and warm crane cabs in 221-F have increased potential for exposure to radioactive materials. These areas are the major concern because the majority of the process is isolated and run remotely due to the high radiation fields.

Internal exposure - The majority of the dose from an intake of material from the Head End Stream comes from plutonium, cerium, and ruthenium, but because different radionuclides target different organs in the body, none of the radionuclides listed in Table A-4 should be ignored. Intakes of the fission and activation products would be detectable with good sensitivity by whole body counts, except for ^{90}Sr and ^{91}Y (the progeny of ^{91}Sr). Not listed, but undoubtedly present, was ^{89}Sr. However, ^{89}Sr would have been counted as part of the gross beta count for ^{90}Sr, and assigning a radiostrontium gross beta result to ^{90}Sr is claimant favorable. If fission products were detected by whole body counts and no urinalysis for ^{90}Sr was available, an intake of 90Sr equal to the intake of ^{137}Cs should be made. Because

^{137}Cs is generally more mobile in the reactor and Head End work environment, this assumption should be conservative. This assumption would account for the dose from ^{91}Sr/^{91}Y as well, which because of the short physical half-lives, produce negligible dose compared to the 90Sr.

Some fission products will be more volatile than others during the dissolution so hopefully whole body counts will provide the ratios of the gamma-emitting radionuclides in the body. If fission products are detected in a worker from this facility, and no 90Sr or plutonium bioassay data are available, assume the ratios provided in Table A-4. An intake of 241Pu using the ratio to 239Pu provided in Table A-3 or for standard 6% or 12% plutonium mixture (Table 4.1.1-4) should also be included. To be claimant-favorable, an intake of 91Y the same magnitude as 106Ru should be assumed.

In the second uranium cycle stream (Stream 2) the majority of the dose comes from ^{238}U and ^{239}Pu; however, the same considerations concerning fission products and ^{241}Pu stated in the preceding paragraph should be applied.

The third stream is known as the second cycle plutonium stream (Stream 3) in which plutonium dominates.

Personnel in this facility are exposed to all processes. Since that is the case, it is not possible to determine the uranium ratios required to determine the Minimum Detectable Intake (MDI). Since ^{238}U is the main isotope in every process stream, the MDI was figured specifically for ^{238}U.

External exposure - Fission products are present in the 221-F Canyon Facility and, depending on the age of the material (i.e., the length of time out of the reactor), the isotopes of concern and the fraction of the activity of these materials will change.

The majority of the external dose from Stream 1 comes from cerium, zirconium, and ruthenium, all photon and beta emitters. In the second stream ^{238}U, which feeds ^{234m}Pa, makes up the majority of the external dose along with zirconium which emits both photons and betas. In the third stream the majority of the external dose comes from ^{241}Pu and ^{95}Zr.

A-5 New Special Recovery (NSR) Facility

The NSR facility recovers plutonium from a variety of sources including scrap plutonium from offsite sources and from the F B-Line facility. The processed plutonium is conditioned so that it can be blended back into the plutonium nitrate feed stream to the F B-Line process.

The New Special Recovery Facility consists of the following processes:

- Material Receiving and Assaying, where the material is assayed to determine its plutonium content before any processing begins;
- Feed Preparation, which is dependent on the type of scrap;
- Dissolution and Filtration, which converts the solid feed material into a solution that is suitable for further purification by either anion exchange or by PUREX solvent extraction;

- Anion Exchange Purification, which removes impurities such as fluoride, aluminum, americium, sulfate, and uranium;
- Waste Handling, which comes from sludge, residues, slurries, and from solid materials such as tools, filters, gloves, etc., and
- Analytical and Assay Operations, where samples are analyzed and returned to the process.

The function of the NSR is to recover plutonium from a variety of materials. These materials also contain impurities such as americium and uranium. The isotopic content is not constant in the material or between materials. For example, the unirradiated reactor cores percent weight range from 6% to 12% for ^{240}Pu, 0% to 10% for ^{241}Am, and 60% to 80% for uranium. It is assumed that the feed material is principally enriched uranium.

Internal exposure - The isotopes that deliver the majority of the internal dose are ^{239}Pu, ^{240}Pu, ^{241}Am, ^{235}U, and ^{236}U (assuming enriched uranium).

External exposure - The isotope that delivers the majority of the photon (external) dose is ^{241}Am. Some neutron dose will also be present due to spontaneous fission and an (α, n) reaction with fluoride, if present.

A-6 F-Area Outside Facilities

The F-Area Outside Facilities provide general support, principally to the processing of irradiated fuels and targets in Building 221-F. The term “Outside Facilities” is used to describe a wide variety of processes, utilities, and services that are ancillary to the primary 200-F Area operations. The main processes of radiological concern are as follows:

Water Handling Facilities
: This facility is used to provide process water and acidified water streams for the canyons, retain discard water in holding tanks for analysis, and decant spent solvent from waste water.

Acid Recovery Unit (ARU)
: This facility concentrates nitric acid for reuse. All material in this location can contain radioactive material.

General Purpose Evaporator:
: The General Purpose (GP) evaporators concentrate aqueous waste having activity that is in excess of disposal limits yet is low enough to be evaporated in unshielded equipment.

General Purpose Waste Tanks
: The tanks in this facility are used to collect GP evaporator feed and to receive and dispense evaporator overheads for recycle to the process.

Waste Handling Facilities
: The F-Area Waste Handling facilities are used for storage and transfer of high- and low-activity wastes that are primarily from the Metallurgical Building 235-F, Savannah River Laboratory (SRL) Waste Concentration Building 776-A, and the Production Control Facilities, Building 772-F and 100-Area sources.

Segregated Solvent Facilities
: Segregated Solvent facilities provide solvent purification and tank storage before the solvent is returned to F-Canyon for reuse.

Transfer Tanks

These tanks provide intermediate pumping stations for those solutions transferring from Building 221-F.

Sump Collection Tanks

Two sump collection tanks are located in separate underground vaults. One tank receives radioactive waste condensate, by jet stream, from sumps or tanks related to the canyon air exhaust system and transfers the waste to the hot or warm canyon waste evaporator feed tanks. Discharges to hot canyon cell sumps are first transferred to the second sump collection tank and then to the canyon for analysis and disposition.

Recycle sump

The Recycle Sump collects drainage and overflow from the canyon auxiliary tanks and holds contaminated or recycled liquids.

Auxiliary Systems and Support Facilities

The only auxiliary system and support facility of concern is the plant laundry located in Building 723-F.

Internal exposure - The radionuclides of concern for the ARU, water handling, and the basin transfer tanks are ^{95}Zr, ^{95}Nb, ^{103}Ru and ^{106}Ru.

External exposure - The radionuclides of concern for the ARU, water handling, and the basin transfer tanks are ^{95}Zr, ^{95}Nb, ^{103}Ru, and ^{106}Ru. The primary radionuclides of external dosimetric concern for the GP evaporators, Segregated Plutonium Solvent, and Laboratory Waste are ^{241}Pu, ^{103}Ru, ^{106}Ru, ^{95}Zr, and ^{95}Nb. The radionuclides of concern for the Segregated Uranium Solvent are ^{103}Ru, ^{106}Ru, ^{95}Zr, ^{95}Nb, and ^{238}U which feeds ^{234}Pa.

The radionuclide distribution in the basin transfer tanks is assumed to be similar to the water handling distribution.

A-7 $^{238}PuO_2$ Fuel Form Facility (PuFF) and the $^{238}PuO_2$ Experimental Facility (PEF)

The primary function of the PuFF facility is to produce encapsulated plutonium oxide fuel forms. This is accomplished by converting ^{238}Pu oxide powder into fuel forms for heat sources by powder ceramic and metallurgical process. The final products are compacted PuO_2 fuel shapes. The PEF provides space for the testing of processes to be used in the PuFF facility. The plutonium oxide blends used in the Puff and PEF facilities are very similar.

The PuO_2 is fabricated and encapsulated in six manipulator cells filled with inert gas, three air-filled manipulator cells, five wing cabinets, and one hood.

The fuel pellets are made by hot-pressing a blended PuO_2 shard mixture prepared from calcined Pu oxalate powder received from H B-Line. Feed processing includes oxygen-16 enrichment, ball milling, compaction, granulation, and sintering. After final heat treatment, the fuel pellets are encapsulated in iridium-clad vent sets by tungsten inert gas welding. The conversion of plutonium oxide powder into dense PuO_2 fuel forms is conducted in the PEF.

Internal exposure - The majority of the dose from the PuFF and PEF material comes from plutonium.

External exposure - The majority of the external dose from the PuFF and the PEF comes from the neutrons generated by spontaneous fission of ^{238}Pu and the photon/beta emissions from ^{241}Am, ^{232}Th, and ^{232}Np.

A-8 235-F Vaults

Vaults in the 235-F Facility were built to store plutonium at different production steps prior to completion and were created not only for safety reasons but also to establish safeguards for security purposes. The 235-F facility receives plutonium and plutonium dioxide and fabricates it into a variety of items used in the nuclear program. There are three vaults located in the 235-F facility:

The Plutonium Oxide Vault
: Used to store plutonium oxide powder from the PuFF and PEF, and the Actinide Billet Fabrication Facility. Neptunium-237 powder and billets are also stored in this facility.

The Finished Product Vault
: Used for the storage of uranium and transuranics in powder form like the plutonium oxide vault. In addition, this vault contains a water cooled, shielded storage area for the storage of finished ^{238}Pu products.

The Scrap Vault
: Used to store metal and oxide scraps of uranium, thorium, and plutonium that are in the form of chips, sweepings, powder, blended samples, or metal turnings.

Internal exposure - Radioisotopes that are stored and found in the 235-F vaults are ^{238}Pu, ^{239}Pu, HEU, and ^{237}Np. All are of concern for determining internal dose.

External exposure - Detailed information on the composition of the radionuclides stored and found in these vaults is not available; therefore, the radionuclides listed in the associated table in the Attachment are considered to be pure. The major source of external exposure is ^{238}Pu spontaneous fission neutrons and betas from ^{238}Pu progeny in addition to gammas and X rays.

A-9 772-F and 772-1F Production Control Laboratories

The production control laboratories provide analytical support for the 200-F and 200-H separations processes. Support is also provided to Waste Management, Reactors, and Raw Materials.

The radionuclides of concern are plutonium, uranium, and fission products. The major source of external exposure is ^{238}Pu spontaneous fission neutrons and betas from ^{238}Pu progeny in addition to gamma, X ray photons, and betas from fission products.

A-10 E-Area Solid Waste Disposal Facility (SWDF)

Solid and liquid low-level waste, generated at the SRS and the SRL, and some waste from off-site are stored at the SWDF.

Solid wastes are located above and below ground. The SWDF is located in areas labeled 643-E and 643-7E. The old burial ground is designated 643-E and ceased operations in 1972. Burial operations were phased in during 1969 to 1972 in the present SWDF designated as 643-7E. Most of the solid waste above ground is TRU waste, mainly ^{238}Pu and ^{239}Pu contaminated waste. Contaminated process equipment contained in steel boxes is stored above ground.

Liquid waste is stored in below ground tanks. Thirty two tanks have been used to store liquids. As of 1988, only three tanks contained liquids. Liquid radioactive waste is in the form of degraded organic process solvents, waste oil, scintillation solutions, and contaminated hazardous waste chemicals.

Due to the diverse nature of the radioactive waste, the radionuclide content of each type of waste has not been defined. The radionuclides of highest concentration (Ci/vol) are tritium, cobalt, plutonium, and curium.

External Exposure - The radionuclides of external dosimetric significance are the fission and activation products associated with reactor operations, such as ^{60}Co, Cs, Sr, Ni, Ce, Rh, and Zr.

A-11 Plutonium Storage Facility (PSF)

The PSF is located in the 200-F Separations Area and is used to receive, store, monitor, retrieve, and ship packaged plutonium. Plutonium in solid and powdered form (such as scrap) contained in approved shipping containers may be temporarily stored there prior to shipment.

The Receipts Assay Facility is the primary location for material accountability. Specifically, it is the area used for the unpacking of off-site material. Material from off-site is unpacked in a shielded glovebox, the inner package is removed, and it is passed on to the loading station where the inner package is weighed, labeled and sealed, then sent to the nondestructive assay laboratory.

Internal exposure - The radionuclide of concern in PSF is plutonium. Many different mixtures of plutonium isotopes are possible, but it is assumed that the mixture is either 6% Pu or 12% Pu, the percent referring to the mass percent of ^{240}Pu.

External exposure - The alpha-emitting isotopes of plutonium are not usually an external exposure concern. However, when they exist in a fluoride form, the alpha particles interact with the fluorine by an (α, n) reaction to produce a significant source of neutrons.

Neutrons are also emitted by spontaneous fission along with associated beta and gamma radiations from ^{238}Pu and ^{240}Pu.

Plutonium-241 decays by beta emissions to ^{241}Am which emits a 60 keV photon and X rays with an energy of 17 keV. These photons are the primary external exposure concerns for ^{241}Pu.

All radionuclides identified in Table A-11 are applicable for determining internal and external dose.

A-12 F/H Effluent Treatment Facility (ETF)

The ETF consists of large, impermeable storage basins (with lift stations located in both F and H Areas) and a treatment facility (Building 241-84H located in H Area). ETF is designed to remove hazardous chemical and radioactive contaminants from the 200-Area liquid effluent waste streams and concentrate them for disposal. The effluent processing steps include feed adjustment (which consists of oxidation, grit collection and removal) and pH adjustment; filtration, which consists of three parallel trains with three stages of filtration; organic removal, consisting of mercury removal ion exchange columns (3) in parallel followed by 2 activated carbon beds in series; reverse osmosis, which removes the bulk of the solid material in the 200-Area liquid effluent wastes using a three-stage reverse osmosis process that concentrates dissolved solids; ion exchange, which removes ionic species of radionuclides and mercury; and evaporation (which is fed concentrate from the filtration and reverse osmosis systems, regeneration solution from the ion exchange process, and the cleaning solution from the filtration and the reverse osmosis processes). Evaporator steam condensate is collected in a condensate receiver and pumped to the H-Area segregated water system for analysis. The overheads are transferred to the organic removal feed tank or recycled to the waste water collection tanks for reprocessing. The evaporator bottoms are sent to the H-Area waste tank farm for transfer to the Saltstone Facility.

Internal exposure - The majority of the annual internal dose from the F/H effluent material comes from plutonium, cerium, and ruthenium.

External exposure - The majority of the external dose from the F/H effluent material comes from cerium, cesium, zirconium, strontium, and ruthenium.

A-13 F/H Cooling Water and Retention Basins

The F and H Areas consist of chemical separation plants and appropriate support facilities for the production of purified radionuclides, principally uranium, plutonium, and tritium. The cooling water and retention basins are support facilities for the separations and waste management areas. Low-level radioactive liquid wastes from the separations areas are routed to the basins when activity is detected.

Cooling Water Basins

When radioactivity is encountered in a cooling water system, immediate action is taken to divert the water and isolate the leak. The cooling water system is flushed until the release guidelines are met.

Retention Basins

Both F and H Areas use a retention basin when it is necessary for temporary storage of storm drains that may be contaminated. After sampling and analysis, the water may be further processed, transferred to ETF, or released.

Internal exposure - The majority of the annual internal dose from the F and H Area basins comes from plutonium and uranium.

External exposure - The majority of the external dose from the F Area and H Area basins comes from ^{137}Cs, ^{90}Sr, ^{106}Ru, ^{144}Ce, and ^{238}U feeding ^{234m}Pa.

A-14 F/H Area Tank Farms

High-level liquid radioactive wastes from SRS are received and managed in large underground tanks in the waste tank farms in 241-F and 241-H. The two farms contain 51 large subsurface tanks and related facilities required for safe handling, processing, and temporary retention of liquid wastes. The liquid waste comes primarily from fuel reprocessing operations in F and H Areas with quantities from the Receiving Basin for Off-site Fuel and Resin Regeneration Facility (RBOF-RRF) operations, SRL, and the 100 Areas. Liquid wastes are classified as either high heat waste (HHW) or low heat waste (LHW), depending on whether or not the concentration of radioactivity is such that forced cooling is required to maintain waste temperature within operating guidelines.

F Area and H Area Reprocessing Waste
: Liquid waste from Buildings 221-F and 221-H is made up of many waste streams generated during the recovery and purification of the reprocessing operation, i.e., the HHW from 221-F & H and the LHW from F Area and H Area.

RBOF-RRF
: Waste from RBOF (244-H) and RRF (245-H) is produced during regeneration of ion-exchange beds, backwashing filters, cleaning and handling of fuel and target elements, and similar incidental operations. Radioactivity in these liquids is approximately 400 μCi/gal, but the volume ranges from 1.0 to 2.3 million gal/yr.

SRL and 100-Area Waste
: Batches of liquid waste from SRL and the reactor areas that are not suitable for discharge are delivered to 211-F where the waste is evaporated, adjusted for alkalinity, and transferred to the waste storage tanks. Reactor waste is handled differently if it contains insoluble materials, such as sludges or slurries, at which time it would be unloaded directly to the appropriate tank.

Waste Processing operations
: Liquid waste from separations areas and other locations are routinely processed through a series of operations allowing segregation and consolidation of the waste components prior to interim storage.

Waste Evaporation
: Radioactive waste received in the waste tank facilities is reduced by evaporation, and the concentrated solutions are immobilized by solidification of the residual salts. The evaporator condensate is continuously monitored; and if the radioactivity exceeds1500 dpm/mL gamma, the condensate is diverted and recycled.

The radionuclide contents of the waste depend on the age of the waste and vary with the source stream and prior processing.

Internal exposure - The majority of the annual internal effective dose equivalent in the F Area combined waste tank is delivered by ^{90}Sr, ^{144}Ce, and ^{244}Cm. The majority of the annual internal effective dose equivalent in the H Area combined waste tank is delivered by ^{90}Sr, ^{144}Ce, and ^{238}Pu.

External exposure - The majority of the external dose in the F Area Combined Tank Waste is delivered by ^{90}Sr, ^{144}Ce, ^{137}Cs, and ^{106}Ru. The majority of the external dose in the H Area Combined Tank Waste is delivered by ^{90}Sr, ^{144}Ce, and ^{238}Pu.

A-15 Waste Certification Facility (WCF)

The Waste Certification Facility (WCF), Building 724-8E, located in the 643-E Solid Waste Disposal Facility, is used to certify and package drums of transuranic (TRU) waste. The functions of the WCF are to assay and certify the contents of TRU waste drums and to package, load, and ship the drums. Only physical handling of sealed drums is performed in the WCF.

Typical generic contents of the TRU waste drums include:

Combustibles: paper, gloves, sweepings, cloth rags

Noncombustibles: HEPA filters, crucibles, glassware

Chemicals: evaporated sludge, caustic soda, soda lime, spent resins

Equipment: glove boxes, motors, scales.

Normally the drums contain either ^{238}Pu or ^{239}Pu depending on the source of the TRU waste.

Internal exposure - The radionuclides of concern for internal exposure are the same as for the Solid Waste Disposal facility, 643-E, and also include tritium, curium, plutonium, and activation products.

External exposure - The radionuclides of concern for external dosimetric purposes are the same as those for the Solid Waste Disposal facility, 643-E and also include curium, fission products, and activation products.

A-16 221-H B-Line Facility

The new HB Line was designed to replace the aging HB-Line production facility. The HB Line consists of three separate facilities: the Scrap Recovery Facility, Neptunium Oxide Facility, and Plutonium Oxide Facility.

The Scrap Recovery Facility is designed to routinely generate nitrate solutions of ^{238}Pu or $^{235}U/^{239}Pu$ scrap suitable for purification by anion exchange or solvent extraction in the canyon. Scrap is received and dissolver batches prepared, based on assay of the scrap. The solid scrap is dissolved in hot nitric acid containing trace fluoride ion, transferred through a filter bag, collected in a tank, sampled for accountability and process control, then diluted with nitric acid and transferred to the proper canyon vessel as a nitrate solution.

The radionuclides of concern in the scrap material are composed of either ^{238}Pu, in oxide form, or $^{235}U/^{239}Pu$ mixture. The $^{235}U/^{239}Pu$ mixture normally contains greater than 20% enriched uranium.

In the Neptunium Oxide Facility, neptunium nitrate solutions are received from H Canyon and F Canyon and transferred to a receipt tank located in the HB Line. The solutions are

precipitated to yield neptunium and plutonium oxalate. The neptunium and plutonium oxalate cake is then loaded into a furnace and calcined to NpO_2 and PuO_2.

In the H B-Line it is assumed that the NpO_2 is 100% ^{237}Np for purposes of dose calculations.

In the Plutonium Oxide Facility plutonium nitrate solution from H Canyon is converted to plutonium oxide powder by the oxalate precipitation and calcinations method.

A-17 H-Canyon Facility

The H-Canyon building was designed for the separation and recovery of ^{239}Pu and ^{238}U from irradiated natural uranium by the PUREX process. Over the years, other processes have been added. These operations include the processing of irradiated enriched uranium to recover ^{235}U, the processing of irradiated neptunium targets to separate and recover ^{238}Pu and ^{237}Np, and the processing of thorium to recover ^{233}U.

In the uranium cycle, standard enriched uranium fuel is dissolved in nitric acid catalyzed by mercuric nitrate. The dissolved fuel is clarified in head end, and the adjusted solution is fed into the solvent extraction system. Uranium and neptunium are extracted from the fission products and separated from each other in the first cycle. The uranium product solution is transferred out of the building for further processing. The neptunium product solution is transferred to the HB Line for oxide conversion.

In the neptunium cycle, irradiated neptunium targets are dissolved in nitric acid. Plutonium-238 and neptunium are separated from fission products by a series of anion exchange resin columns. The plutonium and neptunium product solutions are then concentrated, precipitated to oxalate, and calcined to oxides. The plutonium oxide is either shipped offsite or sent to Building 235-F for formation into heat sources. The neptunium oxide is sent to Building 235-F for refabrication into billets.

The radionuclides of concern will vary with the age of the material.

Internal exposure - The majority of the annual internal dose for the First Cycle Stream comes from plutonium and cerium. The majority of the annual dose for the Second Uranium Cycle Stream comes from uranium. The majority of the annual dose for the Second Neptunium Cycle Stream comes from plutonium and cerium. The majority of the annual dose for the Work Stream comes from plutonium and cerium.

External exposure - In all streams (i.e., Uranium, Neptunium, and Work Streams), the radionuclides of greatest concern for external exposure are the fission products cesium, cerium, zirconium, ruthenium, and strontium.

A-18 211-H Area Outside Facilities

H-Area Outside Facilities include a number of processes that support the separations function of the 200-H Area. The main processes and services of radiological concern are A-Line, Water Handling Facilities, Acid Recovery Unit (ARU), General Purpose Evaporator, Segregated Solvent Facilities, Transfer Tanks, Sump Collection Tanks, Recycle Sumps, Auxiliary Systems, and Support Facilities.

Many of the processes and services do not have any connection to each other in terms of function in the Outside Facilities. Due to this, it will in some cases be necessary to treat each process separately in terms of radiological protection.

A-Line

The H-Area A-Line receives a dilute aqueous solution of uranyl nitrate hexahydrate (UNH) enriched in ^{235}U from the canyon where it is temporarily stored and then shipped off-site in tank trucks.

Water Handling Facilities

The primary equipment for water handling consists of tanks, skimmers, and coolers, all of which can contain radioactive materials. The facility is operated to provide process water and recycle water for the canyons, retain recycle water prior to analysis to permit disposal, decant and discharge skimmed solvent, and store weak acid feed for ARU.

Acid Recovery Unit

Nitric acid is concentrated in the ARU for reuse.

General Purpose Evaporator

Low-activity aqueous waste is concentrated in the General Purpose (GP) Evaporator reducing the volume prior to sending the waste to the tank farm.

Segregated Solvent Facilities

The final purification and storage of the solvent occurs in the Segregated Solvent Facilities prior to its return to the Building 221 canyons for reuse. Activity in the spent solvent is caused primarily by degradation of n-paraffin and Tri Butyl Phosphate (TBP) when solvent is exposed to high levels of radiation. The degradation produces complex zirconium and ruthenium which subsequently contaminate both solvent and product streams. One of the purposes of the segregated solvent facilities is to remove the degradation end products and radioactive contaminants from the solvent.

Transfer Tanks

Seven transfer tanks provide intermediate pumping stations for solutions transferred from Building 221-H and are located in open basins, each containing a sump and a pump. The basins are considered a place of potential exposure to airborne radioactivity.

Sump Collection Tank

This tank receives radioactive waste condensate by steam jet from sumps and tanks related to the canyon air exhaust system and send it by jet stream to other tanks that feed the hot or warm canyon waste evaporators.

Recycle Sump

This sump collects the drainage and overflow for all 221-H tanks that contain contaminated or recycled liquids.

Recycle Vent System

The recycle vent system provides a means to filter contaminated air from tanks and vessels prior to discharge to the atmosphere.

Auxiliary Systems and Support Facilities

The process well water system is the only system of concern for these facilities. There are two cooling water streams that are of importance, the segregated cooling water stream and the recirculating cooling water stream. Both streams are continuously monitored for radioactivity and can be diverted into retention or seepage basins to prevent a release of radioactive material.

Internal exposure - The radionuclide of concern for the A-Line is enriched uranium: ^{234}U, ^{235}U, ^{236}U, and ^{238}U.

External exposure - The radionuclide of concern for the A-Line is natural uranium and the major contributor to the external dose will be photons and X rays from ^{234}U, ^{235}U, and ^{236}U.

The radionuclides of concern for all other facilities are ^{95}Nb, ^{95}Zr, ^{103}Ru, ^{106}Ru, and ^{137}Cs.

A-19 Receiving Basin for Off-site Fuel (RBOF) and Resin Regeneration Facility (RRF)

The RBOF facility handles casks and fuels of various shapes, sizes, and content. Casks of spent fuel are received and washed in the cask basin to remove any dirt accumulated during transit. After the cask is loaded into the fuel basin and the loaded fuel basket within the cask is removed and transferred to a fuel transfer bucket, the fuel transfer bucket then moves the fuel underwater to either the inspection basin, disassembly basin, repackaging basin, or storage basin. There are facilities for inspecting the fuel by mechanical and optical means, disassembling fuel, cutting fuel by cross cutting or slitting, and weighing fuel before and after packaging or other alterations.

The RRF is capable of regenerating both anion and cation type resins. The mixed resin is first depleted with NaOH and then the cation and anion resins are separated after which they are regenerated, rinsed, and returned to the deionizer. Other operations performed at RRF include the removal and disposal of the filter cake from the portable 100-Area filter, chemical cleaning of the filter stones in the portable 100-Area filter, and chemical cleaning of target slugs prior to processing in Building 232-H.

The irradiated fuel received at RBOF contains uranium and fission/activation products. Since the work is performed underwater, the inert gases that escape the fuel bundles are of primary radiological concern.

In addition to irradiated fuel, RBOF performs other operations such as basin purification, resin regeneration, and target cleaning. It was determined that tritium, fission/activation products, and plutonium could be present during these activities.

Internal exposure - The mixture of cation and anion resins in the RRF contains fission and activation products. The radionuclides of concern are ^{134}Cs, ^{137}Cs, and ^{60}Co.

External exposure - The RRF receives a mixture of anion and cation exchange resins from the basin waste deionizer. The resins contain fission and activation products.

A-20 Tritium Facilities

The Tritium Facility includes Buildings 232-H, 233-H, 234-H, and 238-H. Irradiated lithium targets are processed in the tritium facilities to recover, purify, and package tritium.

The radionuclides of concern for internal exposure are elemental tritium and tritiated water. In addition, the activation product, ^{65}Zn is associated with the processing of

lithium targets in Building 232-H. The activation product ^{65}Zn is assumed to be the primary radionuclide of concern for external exposure at the tritium facilities.

A-21 M Area – Uranium Target Fabrication Facility, 313-M

The Uranium Target Fabrication Facility was designed and built to manufacture aluminum clad targets for irradiation in the SRS reactors.

Depleted uranium metal, received from the Fernald plant, was fabricated into target rods that were used for transmutation of ^{238}U to plutonium.

There is no information on the distribution of radionuclides in the 313-M material, and the mixture given in Table A-21 is assumed.

Internal exposure -The majority of the annual internal dose from this uranium mixture is delivered by ^{238}U.

External exposure - The majority of the external dose from the uranium mixture is delivered by ^{238}U which decays to ^{234mPa} which emits a high energy beta.

A-22 320-M, 322-M, and 341-M

Process descriptions for these facilities are not available. However, target extrusion was conducted in Building 320-M, and Building 322-M was a Chemical and Metallurgical Laboratory. The M-Area facilities processed uranium, lithium, and aluminum into fuel and target components for the nuclear reactors. Processing included aluminum alloy formation, degreasing, etching, metal extrusion, hot-die size bonding, and nickel plating.

Uranium and/or enriched uranium are assumed to have been processed. The assumed distributions for depleted and enriched uranium are given in Table A-22. All radionuclides given in the tables are of concern for both internal and external exposures.

A-23 Fuel Fabrication facility, 321-M

The Fuel Fabrication Facility was designed and built to manufacture aluminum clad fuel elements for irradiation in the SRS reactors. In this process, enriched uranium metal is alloyed with aluminum, in concentrations required for reactor irradiation. The alloy is cast into hollow cylindrical ingots from which pre-extrusion billet cores are machined. These cores are extruded into logs, which are then machined into sections, encased in aluminum, and co-extruded into tubes. Cores for other types of tubes are fabricated and assembled into billets in other SRS facilities (Li-Al in Building 320-M and NpO_2-Al in Building 235-F) before they are received at 321-M. The materials processed in Building 321-M undergo the following processes: charge preparation; U-Al alloy storage; and U-Al casting, core machining, billet assembly, and billet outgassing.

Three radionuclide sources were identified in Building 321-M: uranium, plutonium, and neptunium. The majority of the annual internal dose from the identified uranium mixture is delivered by ^{234}U. Plutonium was also processed in 1980.

The majority of the external dose from the uranium mixture is delivered by ^{234}U, ^{235}U, and ^{236}U.

When neptunium targets are prepared, photoemissions from neptunium and its daughters (i.e., ^{233}Pa) present a much higher exposure potential than is normally encountered with uranium based targets.

A-24 S-Area – Defense Waste Processing Facility (DWPF)

Radioactive wastes are stored in existing H-Area waste tank facilities in the form of settled sludge, supernatant liquid, and salt cake. The insoluble solids (sludge) is chemically treated and washed to remove aluminum in preparation for the feed to DWPF. The aqueous fraction undergoes treatment with sodium tetraphenylborate to precipitate the cesium and potassium salts. Also, a slurry of sodium titanate is added to absorb the soluble strontium and plutonium. The precipitate is washed and concentrated in preparation for feed to the DWPF. The aqueous product is subsequently blended with the formatted sludge feed and adjusted with glass formers. The adjusted feed is vitrified to a borosilicate waste glass in a slurry-fed joule-heated melter. The waste glass is encapsulated in stainless steel canisters and stored onsite in the glass waste storage building.

The radionuclide composition of the incoming DWPF feed stream may vary depending on which tanks of waste are being processed.

Two process streams represent the high and low ends of the range of $^{238}Pu/^{90}Sr$ concentrations.

Internal exposure - The sludge-slurry feed stream (Stream 1) has a specific activity of 56 Ci/gal. The majority of the annual internal dose is delivered by ^{90}Sr and ^{238}Pu.

The precipitate-slurry feed stream (Stream 2) has a specific activity of 37 Ci/gal. The majority of the annual internal dose comes from the ^{137}Cs and ^{144}Ce.

External exposure - The majority of the external dose is delivered by ^{144}Ce, ^{137}Cs, ^{106}Ru, ^{90}Sr, ^{238}Pu, and ^{241}Pu in the sludge-slurry feed (Stream 1) .

The precipitate-slurry feed stream has a specific activity of 37 Ci/gal with the majority of the external dose being delivered by the same radionuclides as from the sludge-slurry stream.

A-25 Z Area

Waste tank supernate (low-level waste) is transferred from H Area to Z Area via pipeline. At Z Area, the salt solution (supernate) is mixed with cement, fly ash, and blast furnace slag; and the resulting grout is pumped to vaults where it sets into a saltstone monolith.

The radionuclide composition of the incoming feed stream may vary depending on which tanks are being processed.

Two process streams have been identified on the basis of which radionuclides would deliver the majority of the dose and which radionuclides could be used for tracers. The process streams are the decontaminated salt solution and the saltstone.

Internal exposure - The majority of the annual internal dose from the decontaminated salt solution is delivered by ^{238}Pu and ^{106}Ru. The majority of the annual dose from the saltstone is delivered by ^{238}Pu and ^{106}Ru.

External exposure - The majority of the external dose from the salt solution and the saltstone is delivered by ^{238}Pu, ^{137}Cs, ^{90}Sr, and ^{106}Ru.

A-26 A Area Building 773-A

The radionuclide distribution found in some of the various sections is not available, and the information provided is based on the type of work performed.

Building 773-A has many processes including the following:

Actinide Technology
: This section provides technical support for the Pu and enriched uranium production facilities; Building 221-F, FB Line, Building 221-H, and HB Line. The material used here should be similar to that found in Building 221-F, FB Line, Building 221-H, and HB Line. The radionuclides of concern would include plutonium, neptunium, americium, and uranium.

Analytical Development
: This section provides analytical support for research programs and conducts research and development in the area of process control and analyzer development. The material used in this section may include fission products, uranium, neptunium, curium, and/or californium.

Defense Waste Processing Technology
: This section provides support and conducts research and development in the area of waste processing. The material used in this section would be similar to the material processed in the S and Z area and should contain principally fission/activation products and plutonium.

Hydrogen Technology
: This section is involved in development of hydride process technology and testing materials for use in corrosive and hydrogen environments. The type of radioactive material used in this section is unknown.

Interim Waste Technology
: This section is developing the technology for the Defense Waste Processing facility (DWPF) and the Effluent Treatment facility (ETF). Work is performed on supernate processing, precipitation of strontium and cesium, and filtration and acid hydrolysis of the resulting sludges. The characterization of waste and the investigation of the treatment of effluents by ion exchange and reverse osmosis are also being developed. The material used in this section is assumed to be similar to the material processed in the S and Z Areas and should contain principally fission/activation products and plutonium.

Environmental Technology
: This section processes environmental samples. The material used in this section may contain ^{129}I and ^{14}C.

Laboratory Services

The areas of primary interest in this section include the High Level Cells and the F-Wing Experimental Area. The High Level Cells provide the shielding and confinement necessary for analysis and testing of highly radioactive material. The F-Wing Experimental Area has research and pilot scale production facilities for work with alpha and neutron emitters such as ^{239}Pu, ^{244}Cm, ^{243}Am, and ^{252}Cf.

Materials Technology

This section consists of a Robotics Development Laboratory, Powder Metallurgy Facility, On-site Uranium Reprocess facility, and the Hot Machine Shop. The purpose of the Powder Metallurgy Facility is to develop a new process for fabricating reactor fuel tube cores. The On-site Uranium Recycle Process facility is a prototype facility to assist in the design and operation of a plant scale Fuel Production Facility. The Hot Machine Shop provides a variety of functions and services including machining uranium metal. The material used in this section may include fission/activation products, tritium, and depleted and/or enriched uranium.

A-27 Buildings 735-A and 735-11A

Buildings 735-A and 735-11A house the Radiological and Environmental Science facilities. It is assumed that mainly low-level environmental samples are present in these facilities.

A-28 Building 776-A Liquid Waste Handling Facility

Located in Building 776-A, this facility collects the aqueous waste from Building 773-A. Facilities in this building are provided to strain solids from the waste streams, collect waste in receiving tanks, sample and adjust the pH of the waste tank contents, and transfer the waste to tank trailers for disposal. The radionuclides are believed to be a composite of all the material that is used in Building 773-A operations.

A-29 D-Area Heavy Water Production and Reprocessing

A heavy water production plant in D Area began operation early (1952) in SRS history to concentrate heavy water from Savannah River water to moderate and cool the Site's reactors. The facility stopped production in 1981 because there was a sufficient supply of heavy water.

2.0 OCCUPATIONAL MEDICAL DOSE

SRS required pre-employment and annual physical examinations as part of their occupational health and safety program. These medical examinations typically included diagnostic chest x-rays. The doses from these diagnostic x-ray procedures depended not only on the characteristics of the x-ray machine and the procedure used, but also on the frequency of the examination.

2.1 Examination Frequencies

The frequency of exams differed significantly over the years. Occupational x-rays may have occurred more frequently than on the schedule indicated in Table 2.01. This information will be recorded in the file provided by DOE. All occupational x-rays in the worker file are to be included in the reconstruction of occupational dose.

Table 2.01 lists the frequencies of chest x-rays for different age groups through the years and also identifies specific groups of workers. Specific organ doses to be attributed for PA chest X rays calculated on the basis of the dose conversion factors found in ICRP Publication 34 are given in Attachment B, Table B.01. Organ doses from lateral chest radiography have been estimated at 2.5 times greater than those from the corresponding PA doses based primarily on the greater mAs exposure per radiograph and the somewhat smaller Source to Skin Distance (SSD). For organs not listed in ICRP Publication 34 but specified in the IREP code, doses were determined by analogy with anatomical location (Table 2.04). Thus IREP code organs in the thoracic cavity, but not mentioned in the ICRP Publication 34 were assigned the same dose as the lungs; doses to the organs in the head and neck were assigned the same dose as the thyroid. The head and neck organ dose estimates (i.e. eye/brain), should be somewhat greater than doses actually incurred (hence claimant favorable), because of geometry considerations and at least in the case of brain, because of attenuation by the bony cranium. To ensure claimant favorability in the view of the variations in organ dose described in ICRP Publication 34 (p. 51), the doses for females, which are slightly higher than those for males, were used.

2.2 Equipment and Techniques

Although the Savannah River Site (SRS) medical practices are assumed to have followed the adoption of standards of radiology practice during the 1930s and 1940s to minimize dose to the patient, there is the potential for significant dose from occupational medical x-ray examinations depending upon the type of equipment, the technique factors, the number of photo-fluoroscopic examinations typical in the early years (Cardarelli 2002) and the number of radiographic examinations. SRS medical records include notations in individual worker files regarding both the date and the purpose of the of x-ray examinations.

X-ray organ dose estimates for occupational x-rays administered at the SRS are made for Type I equipment (used from 1950 through 1970), Type II equipment (used from 1971 to July 1985), Type III equipment (used from August 1985 to April 1999), and Type IV equipment (used from June 1999 through 2002). A description of the x-ray equipment used at the SRS is included in Table 2.02. The specific technique factors for these machines are in Table 2.03. Since no technique factors were identified by the SRS for

Type I equipment, organ dose estimates for Type I were made by multiplying the organ dose estimates from Type II equipment by 2.5. This approach is reasonable when compared to other DOE sites (e. g. Hanford) where more information is available for x-ray equipment from the early time period.

For both posterior/anterior and lateral views, a standard source to image distance (SID) of 72 inches (183 cm) was maintained for all types. Additional information indicated that all of the x-ray machines were single phase and that there was no air gap between the patient and the film. It is assumed that the Type II-IV equipment had at least 2.5 mm Al total filtration (see Table 3.1 of NCRP 102) and therefore a HVL of 3.3 mm Al eq. (see Table B.2 of NCRP 102). Tables B-01 and B-02 list organ doses for PA 14" x 17" chest films.

Table 2.01 Frequency of Occupational Chest X-rays at the SRS

Time Period	Frequency	Comment
1950 – 1988	Annually	All employees.
	Annually (1)	Construction asbestos workers.
1989 – 1993	Annually	Employees over 50 years old.
	Biennially	Employees 40 to 49 years old.
	Every 3 years	Employees under the age of 39.
	Biennially	Asbestos workers.
	Annually	DOE, USFS and WSI employees
1994 – 1998	Biennially	Employees over 50 years old.
	Every 3rd year	Employees 40 to 49 years old.
	Every 5th year	employees under the age of 39
	Biennially	Asbestos workers.
	Annually	DOE, USFS and WSI employees
1995	Biennially	Pensioners with documented uptake of 5 rem
1999 – 2002	Every 5th year	Employees in a health surveillance program
	Annually	DOE, USFS and WSI employees
	Biennially	Asbestos workers.

(1) The annual chest x-ray was a minimum received by construction and asbestos workers. Individual workers may have received chest x-rays as often as every 6 weeks. This would be recorded in the file provided by DOE on the individual worker.

Table 2.02 Description of the X-ray Equipment used at SRS

Technique	Time Period	Equipment
Type I	1950-1970	Dupont 2DC Safety Screens, Dupont Cronex 7 film, Picker x-ray Tube, No grid, Manual/Hand processing, Manual collimator.
Type II	1971-7/1985	Dupont Daylight Hi-Speed Screens, Dupont Cronex 7Film, Picker x-ray Tube, Stationary 12:1 Grid, PAKO 3-Minute Film Processor, Manual Collimator.
Type III	8/1985-5/1999	Dupont Daylight Cronex 10 TL Film, Quanta III Screens, TecRad Manual Collimator, Technomed Reciprocatic Upright Bucky 12:1 Grid, Kodak M6AW 90 sec. x-ray Film Processor, Eureka x-ray Tube.
Type IV	6/1999-2002	AGFA Cronex 10TK Film, AGFA Curix Cassettes-Ortho Regular Screens, AGFA Multiloader Processor, TecRad Manual Collimator, Eureka x-ray Tube Rad 68, Technomed Reciprocatic Upright Bucky 12:1 Grid.

Table 2.03 Technique factors Used for Each Type of X-ray Equipment

Machine	View	Current (mA)	Voltage (kVp)	Exposure Time (sec.)
Type I	Techniques unknown. No records were kept with regard to the technique factors used during this time period. The Radiologic Technologists employed during this period are all deceased; therefore, no known resources exist.			
Type II	PA	300	110-120	1/30
Type III	PA	300	120	1/40
Type IV	PA	300	120	1/40

Note: PA indicates a posterior/anterior view, the average PA chest measures 26 cm.
the average Lat. chest measures 34 cm.

Although the SRS has not provided any information on early medical X rays, there is indication that a mobile X-ray unit was used that may have resulted in higher doses. DPSP-57-01-11 indicates that an evaluation of the mobile X-ray unit was conducted and that full size chest radiographs were recommended to reduce personnel exposures. This implies that higher exposures may have been delivered by this unit, which may have been a photofluorographic machine.

2.3 Organ Dose Calculations

ICRP Publication 34 (1982) provides tables of average absorbed dose (mGy) in selected organs for selected x-ray projections at 1 Gy entrance kerma (i.e., air kerma without backscatter), for selected views (including PA), and for selected beam qualities (i.e., various HVLs). These tables provide the basic dose conversion factors for converting air kerma to organ dose. Air kerma was obtained from Table B.3 of NCRP Report 102 (1989). The average air kerma rates for the different machines are calculated using the

cGy per mAs provided in NCRP No. 102 (1989) for specific voltage, current, phase of the machine, and distance to the film.

Finally, the SRS organ doses are found by multiplying the ICRP 34 organ dose conversion factors by the modified air kerma factors. The resulting SRS X ray organ doses for all machines are in Tables B.01 and B.02. The doses are shown in units of rem, assuming a quality factor of 1.0 for X rays. SRS records will indicate the view and in most cases only one view was taken per medical examination.

The following formulas were used to calculate the doses in Table B.01.

Based on the techniques in Table 2.03, the mAs for the types of equipment were calculated for each view:

Current (mA) x Exposure Time (sec) = Current for View (mAs) (2-1)

Example for Type II PA view: 300 mA x 1/30 s = 10 mAs

The air kerma rate for 120 kVp was determined to be 0.3 cGy per 100 mAs (see Table B.3 of NCRP 102) and the air kerma was calculated after converting the rate to air kerma per mA.

Current for View (mAs) x Corrected Air Kerma Rate (cGy/mAs) = Air Kerma (cGy) (2-2)

Example for Type II PA view: 10 mAs x 0.003 cGy/mAs = 0.03 cGy

The air kerma was corrected for the thickness of the chest (26 cm) and for distance between the chest and the plane of the film (5 cm) to obtain the air kerma at skin entrance.

Air kerma at 183 cm x SID squared ÷ SSD squared = air kerma at skin entrance (2-3)

Example for Type II PA view: $0.03 \text{ cGy} \times (183 \text{ cm})^2 \div (152 \text{ cm})^2 = 0.043 \text{ cGy}$

Air kerma at skin entrance was multiplied by the dose conversion factors in Table A.2 through A.8 of ICRP Publication 34 for PA chest and HVL of 3.5 mm Al eq.

Air Kerma (cGy) x Dose conversion factor = Dose for View (cGy) (2-4)

Example for Type II PA view, dose to thyroid:

0.043 cGy x 62 mGy/Gy x 1 Gy/100cGy x 1 rad/10 mGy = 2.7 E-3 rad

For Type 1 equipment used prior to 1970, no collimation was assumed. Therefore, organs not normally in the primary beam for a PA chest were included in the primary beam by using ICRP 34 organ dose conversion factors for procedures where those organs would normally be included in the primary beam. For example, assuming no collimation, the ovaries would be in the primary beam, and dose conversion factors for the abdomen were used. Abdomen dose conversion factors were used for ovaries, testes, embryo, and their analogues, cervical spine dose conversion factor was used for

the thyroid and its analogues, and the PA skull dose conversion factor was used for the eye/brain.

Table 2.04 Analogues for IREP Organs Not Included in ICRP 34

Anatomical Location	ICRP 34 Reference Organ	IREP Organ Analogues
Thorax	Lung	Thymus Esophagus Stomach Bone Surface Remainder organs
Abdomen	Ovaries	Liver/Gall Bladder Urinary/Bladder Colon/Rectum Uterus
Head and Neck	Thyroid	Eye/Brain

2.4 Uncertainty

Error, defined as deviation from the correct, true or conventionally accepted value of a quantity, and uncertainty, defined in terms of the potential range of a stated, measured, assumed or otherwise determined value of a quantity, provides an indication of the confidence of the dose estimates. Error implies knowledge of what the correct or actual value is, which is, of course, not known. Hence a more appropriate term is uncertainty, which is expressed in terms of a confidence level, e.g. 99% (i.e. that the correct or true value, although not actually known, has a 99% probability of falling within the range cited) and includes both precision or reproducibility of the measurement and accuracy, or how close the measurement or estimate of dose comes to the actual or correct value.

Although in theory a large number of factors can introduce uncertainties or affect the x-ray machine output intensity and dose to the patient, in practice only four factors can be reasonably considered to have an impact on dose uncertainty. These are 1) variation in applied kilovoltage, 2) variation in beam current,3) variation in exposure time, and 4) distance from the patient to the source of the X rays (SSD). The influence of such other factors as use of screens, grids, reciprocity failure, film speed and development, while potentially variable, would not affect the beam output intensity.

For a given set of machine settings and parameters, x-ray output should theoretically be constant and unvarying. However, this is not true in practice; although output is essentially constant unless focal spot loading occurs such as might be the case when the power rating of the machine is exceeded. It is unlikely that power ratings were ever exceeded since so doing would be difficult to achieve in practice and would result in damage to the x-ray tube. However, even with the use of so-called constant voltage transformers to control line voltages, slight variations may occur in line voltage input or other internal voltages, which in turn could alter the kVp of the output beam. In general, for a given kVp setting, variation in kVp falls within ± 5% of the machine setting. Since as noted above beam intensity is approximately proportional to the 1.7 power of the kilovoltage, this translates to an uncertainty of approximately ± 8.7%with respect to output beam intensity in the 100 to 120 kVp range used for diagnostic radiographs at the Savannah River Site . For conservatism, this is rounded up to ± 9%.

Similarly slight variations in tube current are normal; as a tube ages, or heats up from usage, tube current may change and typically will drop. Hence, all other factors remaining constant, beam intensity will be reduced, and in direct proportion to the change in tube current. Typically, the reduction in beam output from current variation is not more than a few per cent under normal operating conditions; large decreases in beam output will be readily detected and result in maintenance on the machine to restore the output, or, as a temporary stopgap measure, increase in the current or kVp to provide the necessary intensity for proper radiography. There is no evidence to suggest that these stopgap measures were ever necessary or applied at the Savannah River Site. For a given kVp setting, output of the beam is a function of the tube current, which in turn is measured by a milli-ammeter on the machine and measures average tube current. The measurement is subject to uncertainties, and in addition there may be minor changes in output as the tube heats up from normal usage. These variations are typically small, and hence uncertainty in beam output attributable to current variation has been estimated at ± 5%.

Another parameter that has potential to affect the dose, perhaps significantly, from a diagnostic radiograph relates to the time of exposure. This can be readily understood by noting that a full wave rectified machine produces 120 pulses per second of x-rays. For an exposure time of 1/20 of a second, only six pulses would result. A small error in the timer that resulted in a change of only ± 1 pulse would correspondingly affect the output by ± 17%; for an exposure time of 1/30 of a second, the change in output corresponding to a deviation of ±1 pulse is ± 25%. Early mechanical timers were notoriously inaccurate, although timer accuracy improved significantly with the introduction of electronic timers. However, once again for conservatism, uncertainty in beam output attributable to timers will be assumed to have an upper limit of + 25%.

The final factor that is likely to affect patient dose relates to distance from the source of the x-rays, which is a determinant of the entrance skin exposure. For a given individual, the SSD will be determined largely by the thickness of the patient, and how accurate the positioning is. For a typical patient, this variation in SSD is estimated at no more than a few centimeters, with an upper limit of perhaps 7.5 cm. Using inverse square, this indicates an uncertainty of ± 10 % from this source.

There are two approaches to determination of the combined uncertainty from the above four potential sources of uncertainty. The first, and most conservative in that it gives the greatest range, would be to assume that the uncertainties are additive, which would give an uncertainty range of up to 9 +5+25+10 = 49. However, a more reasonable approach would be to assume that the uncertainties are in fact random, and to compute the statistical root mean square (RMS) value. The RMS value is simply the square root of the sum of the squares, and computes as ± 28.7%. Thus, for any individual entrance skin exposure (ESE) or derived organ dose, an uncertainty of ± 30 % at the 99 % confidence level may be assumed; for further conservatism it may be appropriate to assume that errors are all positive, and only the + 30% should be used.

3.0 OCCUPATIONAL ENVIRONMENTAL DOSE

The occupational environmental dose refers to the dose received by workers outside of the facilities. These doses can be internal and external depending on the characteristics of the individual radionuclides. While most radionuclides when inhaled would represent a dose to various organs in the body, the noble gases would only represent an external dose. These radionuclides are addressed in the following sections.

3.1 Internal Dose from On-Site Atmospheric Radionuclide Concentrations

The internal dose for workers outside of the facilities is determined from the air concentrations resulting from the individual facility releases, from ground level releases and from the resuspension of radioactive materials in the soil.

Unmonitored workers can be exposed to occupational doses internally from on-site releases into the air, from the resuspension of radioactive materials in soil, and externally from ambient radiation and releases of radioactive noble gases into air.

3.1.1 On-Site Releases to Air

Although the annual and semiannual environmental reports contain air monitoring data, these data were determined to be of limited value because of the confounding effects of nuclear weapons testing, lack of spatial and radionuclide resolution, and insensitivity of sampling methods to radionuclides important to dose. Therefore, methods were developed to use well- documented source terms developed by others (Cummins 1991; CDC 2001), coupled with historical meteorology, to estimate radionuclide-specific airborne concentrations for ^{3}H, ^{131}I, ^{41}Ar, $^{238}Pu/^{239}Pu/^{240}Pu$, and $^{234}U/^{235}U/^{238}U$. These radionuclides were determined to account for more than 95 percent of the potential missed dose from inhalation and submersion pathways. Air concentration calculations included spatial and release height considerations for source terms from two processing areas (F- and H-Areas), the heavy water extraction area (D-Area), and five reactor areas (C-, K-, L-, P-, and R-Areas). The 50^{th} and 95^{th} percentile air concentrations were estimated for the eight source term areas, seven other populated areas (shown in Figure 3.1.1-1), and a site average. These calculated air concentrations were the used to derive annual intakes based on an assumed individual ventilation rate of 2,400 cubic meters per year. In addition, the geometric standard deviations (GSD) of these intakes were estimated based on the 95^{th} percentile source terms (Rollins 2003) using the following relationship.

$$\mathrm{GSD} = \left(\frac{95\,\mathrm{th\ percentile}}{50\,\mathrm{th\ percentile}}\right)^{\left(\frac{1}{1.65585}\right)} \qquad (3\text{-}1)$$

These annual intakes and GSDs, by year of operation, are presented in Attachment C in Tables C-01 through C-17.

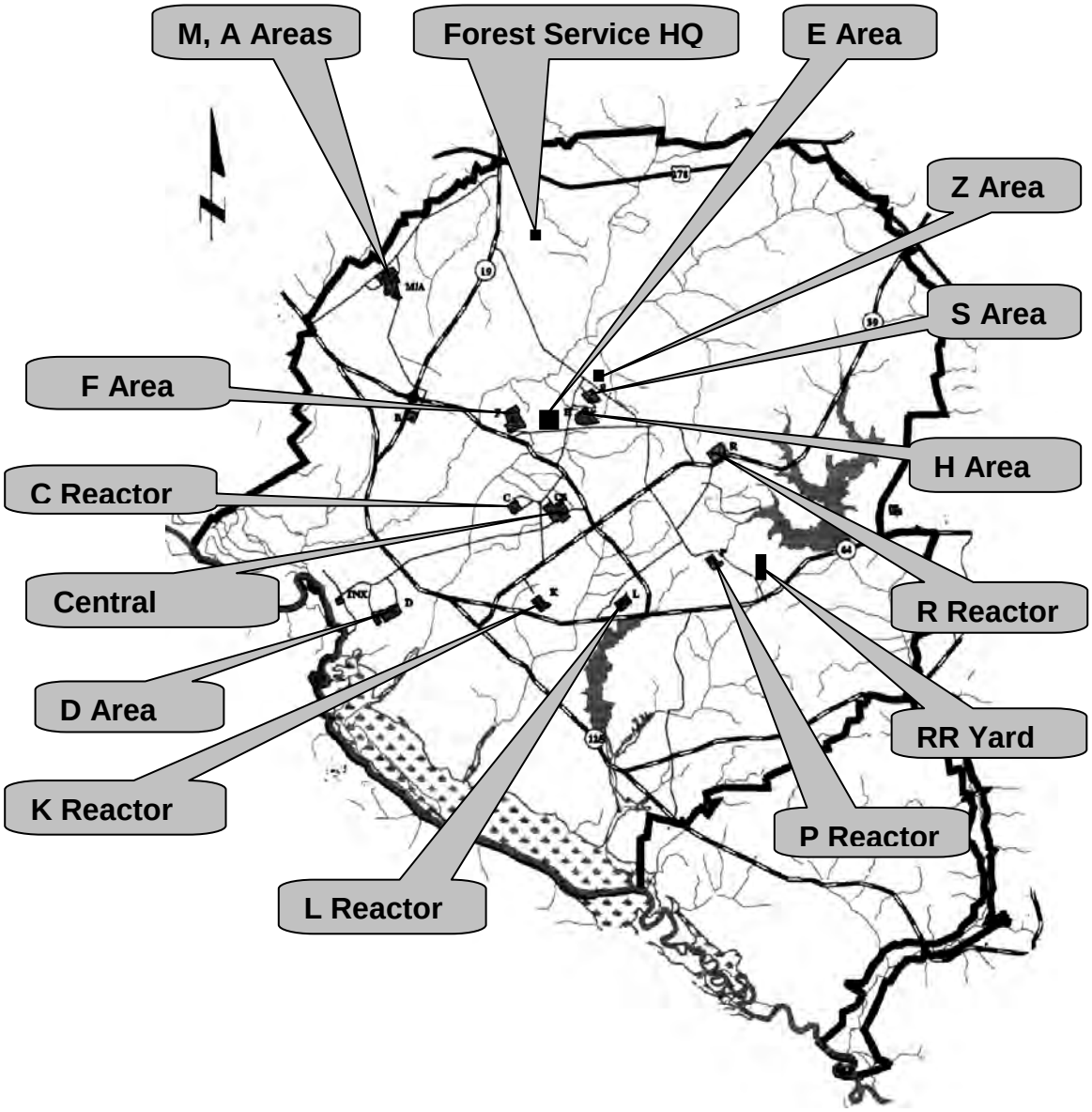

Figure 3.1.1-1 The SRS showing the reactor (C-, K-, L-, P-. and R-Reactors), and processing areas (F Area and H Area) in a rough circle toward the center of the Site, and D Area and A and M Areas near the Site perimeter. Also shown are DWPF (S Area), Saltstone (Z Area), Central Shops, Heavy Water Extraction (D Area), U.S. Forest Service Headquarters, and the Railroad Classification Yard.

Atmospheric dispersion was modeled by developing atmospheric dispersion (X/Q) values using 5-year average joint frequency distributions presented in the SRS environmental reports between 1975 and 1999 (see reference section). These data were converted to X/Q tables for elevated and ground-level releases, using the Multimedia Environmental Pollutant Assessment System (MEPAS) 4.0 computer code (PNNL 1996) and plotted using an Excel® spreadsheet. To assure that atmospheric

dispersion would not be overestimated and resultant air concentrations underestimated, the highest X/Q values in any direction over the 25-year period were chosen as representative for all years of operation. Studies performed at the SRS have indicated that the use of the highest X/Q values would not significantly over or under estimate actual atmospheric dispersion for all years of operation (Harvey 2001, Hamby 1991). Expressions were developed for ground level and elevated releases and used to estimate atmospheric dispersion based on distance from the source. These expressions were developed using the Excel® spreadsheet for all source-term locations (reactor and separations areas) from 500 m out to a distance of 80 km.

For ground level releases (5 m):

$$Y = 1.0146\ X^{-1.8809} \quad (3\text{-}2)$$

For elevated releases (60 m):

$$Y = 0.0013X^{-1.1739} \quad (3\text{-}3)$$

Where

Y = Atmospheric dispersion coefficient, s per m^3
X = Distance from source in meters.

Distances between the populated areas were estimated using geographical information provided by Garmin (2001) and are presented in Table 3.1.1-1. These distances were used in the X/Q expressions and coupled with the adjusted source terms to estimate cumulative air concentrations for all 15 population areas evaluated (Rollins 2003 - TBD).

Table 3.1.1-1 Distance, in Meters, from Source Areas to Population Centers at SRS.

Source Location	A Area	C Area	D Area	F Area	E Area	H Area	S Area	Z Area	R Area	RR Yard	P Area	L Area	K Area	Forest Service	CS
C Area	11500	500	7500	3800	4100	5400	6600	7500	9400	12500	9400	6600	4300	13500	2200
F-Area	8100	4000	10600	500	1600	3500	3900	4700	9100	13900	11300	9700	8300	9700	6500
H Area	10500	5500	12900	3500	1900	500	1200	2000	5700	11100	8900	8800	8900	8800	6400
K Area	15800	4300	6300	8200	8200	8800	10100	10800	10500	11300	8200	3900	500	17500	6900
L Area	17800	6600	10150	9600	8900	8600	10100	10500	8100	7400	4400	500	3900	17600	7600
P Area	19100	9400	14400	11100	9900	8800	9900	10100	5300	3100	500	4400	8100	16900	9600
R Area	15900	9400	16300	9100	7500	5700	6200	5900	500	6100	5300	8300	10500	12000	9400

For radionuclides potentially important for internal dose reconstruction, the average and maximum annual intakes and respective GSDs for all populated areas are provided in Tables C-16 and C-17, respectively. Area-specific annual intakes with respective GSDs are provided in Attachment C (Tables C-01 through C-15) for use where more detailed information is required. The values in these tables were derived from calculated concentrations for the major populated site locations based on historical on-site meteorology, distance from release points, and emissions data.

The values in Tables C-01 through C-15 are estimates of annual intakes derived from the calculated outdoor air concentrations at SRS locations and, as such, may not be appropriate for calculating organ doses for workers located inside buildings. However, these values likely represent the upper bound of potential intakes for an individual working in outside areas where individual monitoring was not required. It is assumed

that these intakes occurred uniformly in the year listed, so for partial years of employment, the intake may be proportionally adjusted by the fraction of the year worked. In addition, these intakes were derived based on an assumed ventilation rate of 2,400 cubic meters per year. Therefore, adjustments may be required based on the job classification depending on the individual performing light work or heavy work for which the respiration rate would be 1.3 m^3/h or 1.7 m^3/h respectively.

For purposes of dose estimation, radioactive decay was not considered. This is considered a claimant-favorable assumption because it would tend to overestimate the actual air concentrations and, thus, the resultant radiation dose. (see Section 3.3 for other uncertainty discussions

In performing dose calculations for unmonitored individuals, three situations may arise;

1. If a worker spent significant time in areas for which radionuclide intakes were calculated and the worker was unmonitored the data in Tables C-01 through C-15 may be used to estimate intakes. These tables show the maximum annual intakes at the 50th percentile and respective GSDs for ^{3}H, ^{131}I, Pu, and U, in each SRS area by year.

2. If the individual worked in multiple areas on the site, the site average annual intakes in Table C-16 should be used. Table C-16 lists the site-wide average annual intakes, which is recommended for performing dose reconstructions to reflect the most likely annual intake to an unmonitored individual. For uncertainty estimation, the GSD (based on the 95th percentile radionuclide source term may be used to define the lognormal distribution.

3. Table C-17 shows the maximum annual intakes of these radionuclides at any SRS area by year. To perform internal dose reconstructions for unmonitored individuals when a specific work location cannot be identified for an individual, maximum annual intakes in Table C-17 (derived from Tables C-01 through C-15 in Attachment C) for each year of employment can be used to estimate the dose to the organ of interest for radionuclides that are most important to dose (Rollins 2003)**.** The use of the maximum values would represent a claimant-favorable dose. The decision to use these maximum annual intakes to estimate unmonitored organ doses should be made cautiously because the worker's intake could be significantly overestimated

These intakes are to be used by internal dose reconstructors along with appropriate values for breathing rate to derive the annual intakes for the significant dose-contributing radionuclides at SRS. Annual intakes provided in Table C-01 through C-17 were derived from the calculated air concentrations assuming a ventilation rate of 2,400 cubic meters per year. These intakes apply uniformly to the year listed, so for partial years of employment, the intake may be proportionally adjusted by the fraction of the year worked.

The following sections discuss details of specific radionuclides of importance for calculations of airborne concentrations

Tritium

Tritium is one of the principal nuclear materials produced at the SRS. It is the heaviest and only radioactive isotope of hydrogen with a physical half-life of 12.5 years. Tritium processing operations have been performed at the SRS facility since October 1955, beginning in Building 232-F. In July 1957, the process was moved to Building 232-H and operations doubled by 1958.

Tritium is produced naturally by cosmic ray interactions and as a fission product in nuclear reactors. The nuclear reactions that produce tritium are:

- Fission of lithium by neutron irradiation of targets
- Reaction of neutrons with the heavy water moderator used in reactors
- Ternary fission of transuranic elements in the reactor fuel and targets.

At SRS, tritium was originally produced as a by-product of producing plutonium in the reactors. Lithium-aluminum control rods were used in R-Reactor and other reactors from the first day of operation to produce tritium and control reactor power. When additional tritium was needed beginning in 1955, enriched uranium was used in the reactors, which allowed for increased loading of lithium-aluminum targets for tritium production.

Gaseous tritium at room temperature tends to form elemental tritium (HT) by reaction with gaseous hydrogen. Tritium closely follows the reactions of ordinary hydrogen when it exists as tritiated water or vapor (T_2O); tritium oxide [HTO]) is taken into the body quite easily, resulting in a higher factor of risk. Tritium is released from SRS in both elemental and oxide forms.

The key processes that have led to tritium releases at the SRS are:

- Reactor operations (100 Areas)
- Recovery of transuranic elements in the separations facilities (200 Areas)
- Recovery of tritium in the tritium processing facilities (200 Areas)
- Laboratory research area (700 Area)
- Heavy water rework facility. (400 Area)

The majority of the tritium releases came from the reactor facilities (100 Areas) and the tritium processing facilities (200 Areas). For purposes of on-site dose reconstruction, only releases from these areas are considered because these releases account for the vast majority of the total tritium released to the atmosphere. Additional information, including histories and descriptions of operations at these facilities and the reconstructed atmospheric tritium source terms from 1955 through 1992, can be found in Savannah River Site Dose Reconstruction Project Phase II: Source Term Calculation and Ingestion Pathway Data Retrieval, Evaluation of Materials Released from the Savannah River Site, RAC Report No. 1-CDC-SRS-1999 (CDC 2001, Chapter 4.1)(hereinafter referred to as the RAC Report). Source terms from 1993 through 2001 were obtained from the SRS annual environmental reports (see reference section for citations). The 50^{th} percentile values were assumed to be those presented in the annual environmental reports. The 95^{th} percentile values were estimated by dividing the 50^{th} percentile yearly value by the smallest value of the ratio of the 50^{th} to the 95^{th} percentile values for all of the years reported in the RAC Report (CDC 2001). This method provides reasonable assurance

that the 95th percentile values for 1993 through 2001 are not underestimated (see Rollins (2003) for details).

The uncertainty analysis applied to reported releases is described in Attachment E of the RAC Report (CDC 2001). The adjusted tritium source terms and uncertainty analyses (CDC 2001, Chapter 4.1) used to calculate atmospheric concentrations attempted to account for the insensitivity of the historical sampling methods and their inability to remove tritium oxide and elemental tritium effectively from the atmosphere. Table 3.1.1-2 presents a description of the monitoring equipment used to monitor atmospheric tritium releases.

Table 3.1.1-2 Overview of Tritium Monitors at SRS (Table E-1, CDC 2001, p. E-2)

Monitor	Facility	Time period in use
Dehumidifier	Reactors	1954–shutdown
Silica gel	Reactors and disassembly areas	during 1954–1958 or until shutdown in some cases
Kanne ionization chambers	Tritium facilities	1954–present
Stack tritium monitor (STM)	Reactors	1970–1988
Berthold tritium monitor (BTM)	Reactors	1988–present
Stack monitor integrator (SMI)	Tritium facilities	1974–present
FORMS (STM with dehumidifier)	Tritium facilities and reactors	1985–present

In the reactor areas, the primary atmospheric release points are the 200-foot stacks and ground-level evaporation from the disassembly basin water. An additional source of atmospheric tritium release is the ground-level evaporation of water purged from the disassembly basin to a seepage basin. Tritium in the form of DTO (from the moderator) is released to the ventilation system and, therefore, to the stack by evaporation of process water exposed to air flowing through the process area. During reactor operations, D_2O containing DTO is evaporated from small leaks in pipe flanges, valves, and from the D_2O process water exposed in the control rod guide tubes. To account for the ground-level releases, the conservative assumption was made (based on The RAC Report (CDC 2001, p. 4.1-8)) that 8% of the total tritium released from the reactor areas was from ground level. This additional amount was added to the adjusted source terms reported in the RAC Report (CDC 2001) and modeled as ground-level release. All releases from the processing facilities were modeled as elevated (200-foot) releases.

Although between 30% and 50% of all tritium released from SRS is thought to be of elemental form (RAC Report (CDC 2001, pp. 4.1-15, 16)), the annual intakes presented in Tables C-01 through C-15 did not attempt to account for this. Therefore, the dose reconstructors should assume that the annual intakes consist entirely of the more biologically effective form of tritium oxide.

Radioiodine

This section discusses the releases of the most important radioiodine isotopes (^{131}I and ^{129}I) to the atmosphere from facilities at the SRS. The historic measurements at the facilities were interpreted in the context of current knowledge of radioiodine chemistry and the effectiveness of various sampling techniques (RAC Report (CDC 2001, Chapter 4.2-2)). These analyses that considered the chemical form, sample line losses, and efficiency of sampling methods resulted in revised estimates of radioiodine releases. These revised estimates, or radioiodine source-terms, were used to calculate air concentrations and annual intakes at 15 populated areas within the SRS. A detailed discussion of methods used to develop the revised source terms and the uncertainty analyses used to develop the 95th percentile values is provided in the RAC Report (CDC 2001, Chapter 4.2). These 95th percentile values were used to derive the geometric standard deviations shown in Tables C-01 through C-17. An overview of the radioiodine sampling and monitoring methods used at the SRS is provided in Table 3.1.1-3.

Table 3.1.1-3 Overview of Radioiodine Sampling and Monitoring Methods (RAC Report (CDC 2001, Table 4.2-2))

Period	Method of sample collection	Comments on reported results
Dec 1954 to Oct 1956	NaOH scrubber	Only reactive forms were measured; system was not considered reliable; no corrections for line loss were made
Oct 1956 to Sept 1961	$AgNO_3$-impregnated filters	Only reactive forms were measured; unknown variability in collection efficiency; no corrections for line loss were made
Sept 1961 to Sept 1965	Charcoal cartridges	All chemical forms were measured; attention paid to collection efficiency variations; no corrections for line loss were made
After Sept 1965	Charcoal cartridges	New monitoring system with higher flow rate; all chemical forms were measured; attention paid to performance; no corrections for line loss were made

Revised radionuclide source terms were developed for both organic and inorganic chemical forms. Although organic radioiodines comprised a large fraction of the total radioiodine released to the atmosphere, for purposes of on-site dose reconstruction, the calculated annual intakes presented in Tables C-01 through C-17 are based on total radioiodine released without distinction to chemical form. This assumption is deemed appropriate because the major differences in the biological effectiveness of the chemical form occur primarily in the air-grass-cow-milk pathway, which is not considered for on-site dose reconstruction.

The fuel processing facilities in F Area and H Area were the sources of the highest releases of radioiodines, which were discharged through tall (60-m) stacks. However, for completeness, elevated releases of radioiodines from the reactor areas were also included in the air concentration calculations and resultant annual intakes. The largest reported releases occurred in 1956. Releases during the years 1955–1961 were significantly higher than the releases that occurred between 1962 and 1971. Those

latter releases were, in turn, markedly greater than releases in the later 1970s and 1980s.

The revised source terms from the processing facilities were provided for the combined releases from F and H Areas (RAC Report (CDC 2001, Chapter 4.2)). To improve spatial resolution of the calculated annual intakes, these combined source terms were partitioned into individual area releases by comparing each area's uranium through-put to the total uranium through-put (RAC Report (CDC 2001, Chapter 4.2 data files)) for both reprocessing areas.

Similarly, revised source terms were provided for combined releases from the five reactor areas. Since there were no quantitative measurements of iodine releases to the atmosphere from the reactor areas prior to 1972, revised estimates were developed based on radioiodine measurements in reactor liquid effluents and later measurement of ratios of atmospheric to liquid releases. To improve spatial resolution of the on-site calculated air concentrations and resultant annual intakes, individual reactor atmospheric releases were estimated based on ratios of reactor-specific radioiodine liquid releases to the combined source terms for all reactors.

Annual intakes of ^{129}I were not calculated because the quantity of ^{129}I released to the atmosphere was several orders of magnitude lower than that of ^{131}I and therefore not deemed important to on-site dose reconstruction.

Plutonium and Uranium

Plutonium and uranium releases to the atmosphere represent the greatest potential radiological risk of all the alpha-emitting radionuclides in the SRS environment. Although releases of these radionuclides were reported for the reactor areas, M Area facilities, and the Administration Area (A Area), the majority of the releases were reported for the two chemical separations areas canyon facilities (221-F and 221-H). Because releases of plutonium and uranium from F- and H- Areas represent more than 97% of the total atmospheric releases of all alpha-emitting radionuclides, air concentrations and resultant annual intakes were calculated only for uranium and plutonium released from the chemical separations areas. The slight underestimation of air concentrations and potential doses resulting from this simplifying assumption is considered to be well within other identified uncertainties.

In general, from 1955 through 1989, the source terms reported by Cummins (1991) for plutonium and uranium were determined to be the best available data (RAC Report 2001 (CDC 2001)). Uncertainties were estimated (RAC Report (CDC 2001, Chapter 4.4)) based on sample collection and counting procedures, and estimated possible sample losses during transmission through sampling probes and lines. These reported values and their estimated uncertainties were used to calculate the 50th percentile annual intakes and geometric standard deviations presented in Tables C-01 through C-17.

Source terms from 1990 through 2001 were obtained from the SRS annual environmental reports (see reference section for citations). The 50th percentile values were assumed to be those presented in the annual environmental reports. The 95th percentile values were estimated by dividing the 50th percentile yearly value by the smallest value of the ratio of the 50th to the 95th percentile values for all of the years

reported in the RAC Report (CDC 2001). This method provides reasonable assurance that the 95th percentile values for 1989 through 2001 are not underestimated.

To simplify the analysis and because isotopic data are limited, calculated annual intakes were developed for total releases of uranium and plutonium. Isotopes assumed to be present in the mix include ^{238}Pu/^{239}Pu/^{240}Pu and ^{234}U/^{235}U/^{238}U. For purposes of dose reconstruction, the isotope of plutonium and uranium contained in the mix that results in the maximum organ dose should be selected and assumed to be present at 100%. This assumption will result in a small, but claimant-favorable, overestimation of the actual dose.

3.1.2 Annual intakes from Resuspension

Soil sampling and analysis were not routinely performed at the Savannah River Site during the period of greatest atmospheric releases (from 1955 through the late 1960s). However, several special surveys were performed between 1958 and 1970, but methods for soil sampling and analysis were not standardized throughout the DOE weapons complex until the early 1970s (RAC Report (CDC 2001, Chapter 12.2, p. 12.2-1)). As of 1973, a laboratory had been dedicated to soils analysis and the necessary field and laboratory equipment had been purchased or fabricated.

Soil analyses conducted in the 1970s and 1980s indicated no measurable SRS-released plutonium at off-site locations. However, elevated levels of certain radionuclides, including plutonium, were detected at several locations near the release points in both F- and H-Areas. Careful review of these data over several years concluded that the measured soil concentrations in F and H Areas were due almost entirely to deposition during the years of the greatest releases. Since the great majority of ^{239}Pu was released from F Area in a single year, 1955, and the majority of ^{238}Pu was released from H Area in 1969, the measured soil concentrations for ^{239}Pu could be conservatively assumed to have persisted in F Area since 1955 and for ^{238}Pu in H Area since 1969 (RAC Report (CDC 2001, p. 12.2-7)). Because the atmospheric releases of nonvolatile radionuclides from other on-site locations were much less than those from F and H Areas, only the soil measurements in F and H Area were deemed important for purposes of on-site dose reconstruction.

For purposes of reconstructing potential missed or unmonitored dose, the annual intakes provided in Table C-18 should be used only for individuals whose employment records indicate that they spent considerable time in either F or H Areas. The ^{239}Pu annual intakes should be assumed to have persisted since 1955, and the ^{238}Pu annual intakes since 1969. Annual intakes resulting from concentrations of ^{90}Sr and ^{137}Cs should be conservatively assumed to have persisted essentially unchanged since 1955.

Section 1.3 above discusses available information regarding chemical forms of radioactive materials in use at SRS for major facilities and processes. These chemical forms, and the SRS assumptions regarding ICRP 30 inhalation solubility classes, provide information to assist the internal dose reconstructor in determining appropriate solubility parameters for use with the IMBA program.

Additionally, Attachment A contains a number of tables providing information related to radionuclide mixtures and dosimetric quantities assumed to be applicable for those radionuclides as taken from SRS program documentation. It is important to note that these dosimetric parameters were based on ICRP 30 recommendations. Internal doses reconstructed in support of EEOICPA are derived using more recent ICRP 68 (1994) recommended methodology than was in use at SRS. Consequently, inhalation solubility classes, dose conversion factors, and annual dose fractions discussed in Section 1.3 and Attachment A of this document should guide the selection of similar quantities used for EEOICPA internal dose reconstructions.

Table 3.1.2-1 Radionuclide Soil Concentrations, pCi g-1, (1 sigma) Dry Weight (0-8 cm depth)

Location	Sr-90	σ	Cs-137	σ	Pu-238	σ	Pu-239	σ
F-Area	0.0216	0.0182	1.05	0.05	0.403	0.014	0.531	0.014
H-Area	0.0289	0.0183	1.32	0.05	0.0213	0.0016	0.0554	0.0023

Source: CDC 2001, Table 12.2-8, on p. 12.2-8, Arnett 1993

Annual intakes resulting from resuspension of radionuclides in soil in concentrations given in Table 3.1.2-1 are presented in Table C-18 and were calculated using the following formula:

$$CA = SD \times CS \times SF \times RF \quad (3\text{-}3)$$

where
CA = Air concentration, pCi m^{-3}
SD = Soil density = 1600 kg m^{-3} (PNL-10550, 1995)
CS = Radionuclide soil concentration = pCi kg^{-1}
SF = Surface factor = 0.08 m (Arnett 1993)
RF = Resuspension factor = 1.00E-09 m^{-1} (Till and Meyers, 1983, p. 5-32)

3.2 Occupational External Dose

Workers are subjected to external doses from the ambient radiation levels and from noble gases. Prior to 1966, ambient radiation levels were measured with pocket ionization chambers and GM detectors. After 1966, the measurements were made with ion chambers and thermoluminescent dosimeters (TLDs). Concentrations of the noble gas ^{41}Ar were calculated and can be used to determine the external dose due to submersion.

3.2.1 Ambient Radiation

The environmental radiological profile has been developed for the Savannah River Site for use by dose reconstructors when personal dosimetry or bioassay program participation was not required. Site annual environmental reports were reviewed for data that would be useful in reconstructing ambient radiation levels and airborne radionuclide concentrations. Data in these historical documents (see reference section for citations) included ambient ion chamber and thermoluminescent dosimetry (TLD) radiation measurements, air sampling results, and residual soil activity. The historical ion chamber and TLD results with uncertainty estimates (geometric standard deviations), where available, were tabulated and are presented in Table C-19.

The ambient radiation levels presented in Table C-19 were taken directly from the SRS annual and semiannual environmental reports (see reference section) and include contributions from the SRS releases and background radiation. Until 1966, ambient radiation measurements were performed using either pocket ionization chambers or GM detectors, and results were reported in units of millirep (mrep), milliroentgen (mR), or millirad (mrad) per 24 hours without any estimates of uncertainty. In 1966, the Savannah River Site began performing ambient radiation measurements with TLDs. For purposes of this report, mrep, mR, and mrad were assumed to be equivalent to mrem. Although ambient radiation results were typically reported in 24 hour increments, measurements were usually performed over one-week periods resulting in between 600 and 800 measurements per year from all locations. The values in Table C-19 are for 2,000 hours of exposure per year.

In 1974, the environmental reports began presenting a quantitative estimate of the uncertainty associated with the maximum and minimum radiation measurements. Along with the maximum and minimum radiation values, the reports also provided a value for two sigma. The 50^{th} percentile values presented in the tables are generally the average reported values for a given area. The geometric standard deviation values in Table C-19 were based on the reported two sigma value. It should also be noted that all of these values include the contributions from natural background radiation and nuclear weapons fallout. Therefore, use of these data to account for potential missed or unmonitored dose should in all cases represent an overestimation of the worker's actual dose.

3.2.2 Releases of Noble Gases

In addition to radionuclides important to internal dose reconstruction, ^{41}Ar was released in relatively large quantities during reactor operation The radionuclide ^{41}Ar is a noble (nonreactive) gas with a half-life of 1.8 hours and is produced in the air surrounding the reactor. Because this air volume communicated with air in the reactor building, most of the ^{41}Ar produced in the reactor was released with ventilation air exiting the 200-foot reactor stack. The five reactors were responsible for the vast majority of the atmospheric releases of ^{41}Ar. Neutrons interacting with the naturally occurring ^{40}Ar in the air surrounding the reactor vessel created ^{41}Ar which quickly mixed with reactor building air and was exhausted through the main stack. Once out of the stack, reactor gases moved and were dispersed with local air flows and both on-site and off-site individuals nearby were potentially exposed to beta and gamma radiation emitted during the ^{41}Ar decay.

The most reliable, reactor-specific, source term estimates, according to CDC (2001, p, 4.3-1), were provided by Cummins (1991). These source terms were used to calculate the 50^{th} percentile annual whole body and organ doses presented in Tables C-20 andC-21. Because Cummins (2001) did not include estimates of uncertainty, the 95^{th} percentile source terms were estimated by assuming that the 95^{th} percentile release was simply 25 percent greater than the 50^{th} percentile release.

This potential dose resulting from submersion in the plume of this noble gas is considered to be entirely external because the gas is not readily assimilated by the body. Using methodology identical to that for the radionuclides important to internal dose, calculated air concentrations at the populated areas of the SRS were estimated. These

concentrations were then converted to organ and whole body dose using submersion dose conversion factors published in Federal Guidance Report No. 12 (EPA 1993). The annual doses (assuming 2000 hours per year of exposure) have been tabulated based on 50th percentile source terms presented in Tables C-20 and C-21. The geometric standard deviations provided in these tables were based on the estimated 95th percentile source terms using the relationship provided in Section 3.1.1 above.

For purposes of external dose reconstruction, ^{41}Ar doses to the skin (Table C-20) should be assessed as 30% from >15 keV electrons and 70% from >250 keV photons. Doses to bone surfaces (Tables C-20 and C-21), should be assumed to be 100% from >250 keV photons. In addition, ^{41}Ar dose to whole body and other underlying organs (Tables C-21) should be assumed to be 100% from >250 keV photons.

3.3 Uncertainty

As discussed in the previous sections, estimates of annual intakes employed conservative (i.e., claimant favorable) assumptions. For example, the use of maximum, 25-year, sector X/Q values to estimate atmospheric radionuclide concentrations. Rollins (2003) validated that these calculated air concentrations would likely overestimate the actual radionuclide concentrations by comparison to actual measured data taken from the SRS annual environmental reports. It should be noted that the estimated annual intakes were based on annual average meteorology and that the actual instantaneous meteorology on any given day could vary from these averages by factors (plus or minus) of several orders of magnitude. However, the intake values given in Attachment C should represent a reasonable upper bound of the actual intakes that could have occurred.

In instances where more detailed information is known about a particular individual or job classification, other modifying factors should be employed. For example, if, for a particular job classification, there is reason to believe that the actual ventilation rate for the claimant may vary markedly from the average of 2,400 cubic meters per 2,000 hours of exposure, the dose reconstructor should use professional judgment to adjust the estimated intakes as necessary whether the individual is engaged in light or heavy work. The respiration rate for light work is 1.3 m^3/h and 1.7 m^3/h for heavy work. In these cases, to estimate annual intake, the product of the fractional annual period for each, job dependent, level of work and the corresponding ventilation rate, are summed to determine the total ventilation volume for the year m^3. The annual intake is then determined by multiplying the annual ventilation volume by the annual average concentration in Tables C-1 through C-17 for the year and location of interest.

In general, the annual intake values presented in tables C-1 through C-17 reflect adjusted source terms and uncertainty analyses provided in CDC (2001). For purposes of this document, no other attempt has been made to quantify other uncertainties.

4.0 OCCUPATIONAL INTERNAL DOSE

Radionuclides of concern listed in the SRS Internal Dosimetry Technical Basis Manual (WSRC-IM-90-139, 31 December 2001), chapter 5, include the following elements and isotopes:

- tritium (^{3}H)
- manganese (^{54}Mn)
- cobalt (^{60}Co)
- zinc (^{65}Zn)
- strontium (^{90}Sr)
- ruthenium (^{106}Ru)
- antimony (^{125}Sb)
- cesium (^{134}Cs, ^{137}Cs)
- cerium (^{144}Ce)
- europium (^{152}Eu, ^{154}Eu)
- thorium (^{228}Th, ^{232}Th)
- uranium (^{234}U, ^{235}U, ^{236}U, ^{238}U, and mixtures)
- neptunium (^{237}Np)
- plutonium (^{238}Pu, ^{239}Pu/^{240}Pu, ^{241}Pu)
- americium (^{241}Am, ^{243}Am)
- curium (^{244}Cm)
- californium (^{252}Cf, including ^{248}Cm daughter)

These radionuclides were those identified by SRS as the most significant at delivering doses or that were effective tracers for significant dose-delivering radionuclides. A summary of the data in this section is presented in Attachment D.

4.1 *In vitro* Minimum Detectable Activities (MDAs), Counting Methods

From WSRC-IM-90-139, the decision level for urinalysis is calculated as:

$$D_L = \frac{\frac{4.65}{2} * \sqrt{C_{BACKGROUND}} + 3}{TREVA} \qquad (4\text{-}1)$$

where

D_L is in dpm/L
$C_{BACKGROUND}$ = background counts in the region of interest
T = count time
R = recovery
E = average detector efficiency
V = sample volume (usually 1.5 L), and
A = the alpha abundance for the radionuclide in question.

The decision level is the level at which activity is considered present in a sample with a 95% confidence level.

Similarly, the MDA is calculated as:

$$MDA = \frac{4.65 * \sqrt{C_{BACKGROUND}} + 3}{TREVA} \qquad (4\text{-}2)$$

The *reporting level* is distinguished from the MDA. As described in WSRC-IM-90-139, the reporting level is the minimally acceptable decision level; the bioassay laboratory was required to achieve decision levels below the reporting level. SRS technical documentation provided no quantitative relationship between the reporting level and MDA.

4.1.1 *In vitro* Urine Analysis

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

Current values of MDA for *in vitro* analyses of routine samples are shown in Table 4.1.1-1 below, extracted from WSRC-IM-90-139. The MDA values are 2 times the average decision level for the analysis. These values are applicable from 1988 through present for plutonium, 1990 through present for uranium, and 1994 through present for the trivalent actinides americium, curium, and californium. No specific beginning dates were found for tritium, ^{90}Sr, or ^{237}Np.

These MDAs apply to both routine urinalyses and to special urinalyses. Since about 1990, routine analysis results (except tritium) that exceed the decision level are recounted for twice the normal count time. The recounted result is placed in the database instead of the original count, so the MDA for a recount result will be approximately 0.7 times the values shown below. Some samples are "rerun," meaning a second aliquot is processed. In these cases the database will show two results with different sample numbers for the same collection date and same sample volume.

Table 4.1.1-1 Current MDAs for Urinalysis

Radionuclide	MDA (pCi/L)	Radionuclide	MDA (pCi/L)	Radionuclide	MDA (pCi/L)
H-3	20000	U-234	0.032[(a)]	Pu-239	0.022
Sr-90	2.9	U-235	0.036[(a)]	Pu-240	0.022
		U-238	0.032[(a)]	Am-241	0.029
		Np-237	0.029	Cm-244	0.021
		Pu-238	0.032	Cf-252	0.021

(a) Despite the listed MDAs, uranium results below 0.15 dpm/L are not considered "detected." See the uranium discussion in 4.1.2 for details.

Activity fractions for depleted, recycled, and high enriched uranium are listed in the glossary section.

Analysis methods for these radionuclides are liquid scintillation counting for tritium, beta counting by gas proportional counter for ^{90}Sr, and alpha spectroscopy for the remaining radionuclides in table.

4.1.2 *In vitro* Methods for Individual Radionuclides

The following section discusses the in vitro methods for specific radionuclides.

***In vitro* bioassay for trivalent actinides (americium, curium, and californium)**

Records showing urinalysis for trivalent actinides date back at least to the mid 1960s. The early reporting levels varied from 3 to 1 dpm/1.5 L. In 1971 the reporting level using gross alpha counting on a solid state detector was listed as 0.3 dpm per 1.5 liters. In 1990, a change in radiochemical processing resulted in an MDA of 0.1 dpm per liter. Alpha spectrometry has been used since 1994. For MDAs since 1994, see Table 4.1.1-1 above.

Use of a gross alpha count prior to 1994 means that the measurement itself does not indicate which radionuclide or combination of radionuclides is being measured. Location or other information concerning an intake may allow determination of the principal radionuclide. If not, it is claimant-favorable to assume ^{241}Am.

***In vitro* bioassay for plutonium**

From the beginning of the plutonium urinalysis program in 1954 to approximately 1959, urine samples were radiochemically processed using bismuth phosphate and lanthanum fluoride coprecipitation and electroplated, and activities were determined by gross alpha track analysis of exposed NTA emulsions. In 1959 nitric acid/hydrogen peroxide dissolution and ion exchange replaced the bismuth phosphate method. This was faster and used less volume of urine but had essentially the same MDA. The reporting level did not change. Results were recorded as Pu or sometimes as Pu238/Pu239. From around 1964 to 1988, counting for gross alpha activity was performed using a solid state surface barrier alpha detector. Tri-isooctlyamine (TIOA) liquid extraction replaced the ion exchange chemistry in 1966. This method also used direct evaporation on planchets instead of electrodeposition. This method also allowed separation of neptunium and uranium from the same sample. Sensitivity was stated at 0.1 dpm/1.5L, which was consistent with the reporting level already in use. In or about 1981, a new coprecipitation technique was introduced for routine samples along with alpha spectrometry. Sample-specific determination of plutonium recover by use of a 242Pu tracer was also introduced at that time. Results for ^{238}Pu and ^{239}Pu were reported separately. The TIOA method with gross alpha counting continued to be used on special samples until 1988. An electronic database was introduced in 1990 and results were thereafter reported as per liter. Electrodeposition was reinstated in 1994. Separation of plutonium+neptunium, actinides, uranium, and strontium from a single sample using TEVA and TRU resins began in 2001. Alpha-emitting plutonium and neptunium isotopes are electrodeposited and counted by alpha spectrometry on a single planchet. Reporting levels for the various radiochemical and counting methods are listed in Tables 4.1.1-2 and 4.1.1-3. Note: MDAs are assumed to be more or as sensitive as the reporting levels.

Table 4.1.1-2 Pu-239/240 – in vitro Urinalysis

Time Period	Reporting Level[(a)]	Method
1954-1959	0.05 dpm /1.5L	Bismuth phosphate, lanthanum fluoride coprecipitation; gross alpha for plutonium; alpha track counting
1959-1962	0.05 dpm/1.5L	Ion exchange, gross alpha for plutonium, alpha track counting (until 1964)
1963-1966	0.1 dpm/1.5L (1963)	Solid state alpha counting (1964)
1966-1981	0.1 dpm/1.5L	TIOA extraction; gross alpha for plutonium on solid state detector
1981 – 1988	0.07 dpm/1.5L for routine samples	Alpha spectroscopy for $^{239/240}$Pu (used primarily for routine urine samples)
1988-present	0.06 dpm/L	Alpha spectroscopy for 239/240Pu

Table 4.1.1-3 Pu-238 – in vitro Urinalysis

Time Period	MDA/Reporting Level – Urinalysis	Urine Analysis Method
1954-1981	Reporting levels same as in Table 4.1.1-2, ^{238}Pu and ^{239}Pu combined into single result	Methods same as in Table 4.1.1-2
1981 – 1988	0.05 dpm/1.5 L [reporting level]	Alpha spectroscopy for ^{238}Pu (used primarily for routine urine samples)
1988-present	0.032 pCi/L (~0.07 dpm/L) [MDA]	Alpha spectroscopy for ^{238}Pu

Because plutonium urinalysis results (fecal results as well) prior to 1981 (1988 for special samples) measured the total activity from alpha-emitting isotopes whereas ^{238}Pu and $^{239+240}$Pu results were reported separately from 1981 to the present, the two sets of data cannot be entered into the intake code as reported. If, for a specific case, all the bioassay data are prior to 1981, the total plutonium alpha data can be entered as ^{239}Pu. The calculated intake will have to be partitioned into ^{238}Pu and ^{239}Pu per the mixture ratios applicable to the intake (if known). However, it is unlikely that for the older intakes, intake-specific isotopic mixture ratios were established so the default mixture ratios will have to be used. The same mixture ratio will then be used to determine the intake of ^{241}Pu and ^{241}Am.

Plutonium existed in mixtures of ^{238}Pu, ^{239}Pu, ^{240}Pu, and ^{241}Pu. The activity of ^{242}Pu was always insignificant dosimetrically. The relative activities of these isotopes depended on the nature of the irradiation of the uranium fuel (referred to a burn-up) and the time between the end of irradiation and the intake. Processes at SRS did not perturb the relative activities of the isotopes. Generally, SRS created plutonium mixtures ranging from about 3% per weight ^{240}Pu to about 12% by weight ^{240}Pu. Generally, the plutonium mixtures were blended to produce a final product with about 6% by weight ^{240}Pu, which is referred to as a weapons-grade mixture. The ^{241}Pu, a beta-emitter with a 14-yr half-life, undergoes noticeable decay to ^{241}Am during the lifetime of the facilities so that the isotopic ratios of contamination will be different than the ratios in newly irradiated fuel (or new feed to the Head End Stream). ^{241}Am builds up from near zero at end of irradiation;

however, it is removed during production of the plutonium product, then begins to build up again as the ^{241}Pu remaining the product decays. So the ratio of ^{241}Pu to, say, $^{239+240}$Pu decreases from time of end of irradiation; whereas the ratio of ^{241}Am to $^{239+240}$Pu increases from time of last separation of the ^{241}Am from the plutonium.

If for a given intake the activity of ^{241}Am was established, through chest counting, trivalent actinide excreta analysis, or analysis of the source of the contamination, then that activity should be used. If unknown, the default ratios provided in Table 4.1.1-4 can be used as best fits the nature of the intake. Fresh 12% plutonium mix is the most claimant-favorable.

Table 4.1.1-4 Activity Composition for Reference 6% and 12% ^{240}Pu Mixtures[a]

Isotopic Component	Reference 6% Pu Mix			Reference 12% Pu Mix		
	Fresh[b]	Aged[c] 5 Years	Aged 10 Years	Fresh	Aged 5 Years	Aged 10 Years
Specific Activity in Mixture, Ci/g						
Pu-238	8.56E-3	8.23E-3	7.91E-3	1.71E-2	1.64E-2	1.58E-2
Pu-239+240	7.13E-2	7.13E-2	7.13E-2	7.98E-2	7.98E-2	7.98E-2
Pu-241	8.24E-1	6.48E-1	5.09E-1	3.09	2.43	1.91
Am-241	0	5.83E-3	1.04E-2	0	2.19E-2	3.89E-2
Pu alpha	7.99E-2	7.95E-2	7.92E-2	9.69E-2	9.62E-2	9.56E-2
Total alpha	7.99E-2	8.53E-2	8.96E-2	9.69E-2	1.18E-1	1.35E-1
Activity Ratios						
Pu239+240: Am-241	NA	1.22E1	6.87	NA	3.64	2.05
Pu-alpha: Am-241	NA	1.36E1	7.61	NA	4.39	2.46
Pu-241: Pu-239+240	1.16E1	9.09	7.14	3.87E1	3.05E1	2.39E1
Pu-241: Pu-alpha	1.03E1	8.15	6.43	3.18E1	2.53E1	2.00E1

a. From *Methods and Models of the Hanford Internal Dosimetry Program* (Carbaugh, 2003). Also found in part in *Guide for Good Practices for Occupational Radiological Protection in Plutonium Facilities* (USDOE 1998)
b, Two weeks after end of irradiation
c. Time since separation of the ^{241}Am

If some of the bioassay data were obtained prior to 1981 and some after, then the data will have to be made consistent. This can be done in either of two ways. 1) If all the ^{238}Pu and ^{239}Pu results are above their respective decision levels, then the ^{238}Pu and ^{239}Pu results for each sample can be summed to get a total plutonium alpha result. This result can be equated to the pre-1981 results and treated in the manner described in the preceding paragraph. 2) If some of the ^{238}Pu or ^{239}Pu results are below their decision levels, then it may be best to use the isotope that has the most robust data set and adjust each of the pre-1981 results to reflect just the portion from that chosen isotope. The adjustment will be made using the default isotopic mixture ratios unless the intake-specific ratio is known. For instance a gross alpha result of 2.1 dpm/1.5L would be converted to 1.9 dpm/1.5L of ^{239}Pu before being entered into the intake code.

In vitro bioassay for tritium

From startup until 1958, HTO in urine was analyzed by passing hydrogen evolved from the urine sample through an ionization chamber; the reported MDA for this method was 1 µCi/L. In 1958, liquid scintillation counting was initiated, which is still the method in use. *A History of Personnel Radiation Dosimetry at the Savannah River Site* (WSRC-RP-95-234) reports that the MDA consistently improved to the current level of 20,000 pCi/L (or 0.02 µCi/L). The reporting level remained at the value of 1 µCi $L^{-}1$, and was later reduced to 0.5 µCi L^{-1}, on or about 1981, and then to the current value of 0.1 µCi L^{1}. (an exact date was not determined at least by 1989). During the 1980s, although the reporting level of 0.5 µCi L^{-1} was generally used, some results below 0.5 are listed directly, e.g. 0.4, 0.3. Considering that the true MDA was probably well below the reporting level, these results should be considered as real.

It should be noted that for current analyses, tritium results of 0.05 µCi L^{-1} or less are reported as "<0.1µCi L^{-1}," and results between 0.05 µCi L^{-1} and 0.1 µCi L^{-1} are reported as "0.1 µCi L^{-1}." Results greater than 0.1 µCi L^{-1} are reported as measured (to 1 significant figure).

Tritium analyses are listed as "T" on the employee bioassay cards. Tritium may also be listed as "P-10," especially in the 1950s. Tritium results in the 1990s were listed on the same summary form as external dose monitoring results. They are referred to as sample results with dates and analysis results, but the word "tritium" or any other radionuclide identifier is not mentioned per se.

For tritium results, the denominator used for reporting purposes has always been per *liter* of urine. (The denominator of 1.5 liters was never used for tritium as it was for other radionuclides.)

Only tritium bioassay results greater than 5 µCi L^{-1} were evaluated prior to computer evaluation of data.

In vitro bioassay for uranium

A variety of methods have been used historically to analyze uranium at SRS. These methods and the associated detection capabilities are summarized in Table 4.1.1-5.

Activity fractions for depleted, recycled, and highly enriched uranium are listed in the glossary section. SRS technical documentation indicates that for earlier monitoring periods, the designations "enriched" and "depleted" analysis for uranium referred to analysis performed by alpha counting or chemical measurement, respectively, and was 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 U was used to designate the fluorophotometric method.

Table 4.1.1-5 Uranium Urinalysis

Time Period	Uranium mixture	Reporting Level	Urine analysis method
Startup to mid-1960s	Enriched uranium	0.15 dpm/1.5 L	Gross alpha for uranium, alpha track counting
Mid-1960s to 1982	Enriched uranium	1 dpm/1.5 L	Gross alpha for uranium on solid state detector
1954 - 1982	Depleted uranium	1-5 µg/L	Fluorophotometric analysis
1982 - 1986	U-235	0.14 ng	Delayed neutron analysis (DNA)
	U-nat	1 µg/L	DNA analysis for U-235
	Enriched uranium	1 dpm/L	DNA analysis for U-235
1986-1990	Enriched uranium	1 dpm/1.5 L	Gross alpha for uranium on solid state detector
1986-1994	Depleted uranium	[not listed in WSRC-RP-95-234]	Kinetic phosphorimetry analysis (KPA)
1990 – present (enriched uranium), 1994-present (depleted uranium)	DU	use U-238 and isotopic ratios from glossary	Alpha spectroscopy for specific uranium isotopes
	RU	use U-238 and isotopic ratios from glossary	
	HEU	use U-235 and isotopic ratios from glossary	
	U-234	0.032 pCi/L	
	U-235	0.036 pCi/L	
	U-238	0.032 pCi/L	

Because of excretion of uranium from natural sources, the MDAs for natural or depleted uranium listed above should not be used to determine occupational intakes. Based on a probit analysis of 1728 worker samples in 1993-1994 (WSRC-IM-90-139, rev. 8), an upper limit at the 99% confidence level for analytical noise and presumably some effect of natural background was set at 0.15 dpm/L. The SRS policy is that a sample result must exceed both its decision level and 0.15 dpm/L to be considered a detection. Lacking other information, i.e., similar studies at different times throughout the history of the SRS, it is reasonable to apply the 0.15 dpm/L criterion to all historical samples. The background excretion in terms of µg/L was not established, but almost certainly was below the 1-µg/L reporting level.

If isotopic data are available, they can also be used to determine if a uranium urinalysis result represents an occupational intake. The following rules apply:

- If ^{235}U or ^{236}U are detected, then ^{234}U must also be detected.
- If both ^{234}U and ^{238}U are detected, the uranium should be considered from occupational sources if the $^{234}U/^{238}U$ ratio is outside of a range from 1 to 3.
- If ^{238}U is detected, the uranium should be considered from an occupational source if the ratio of the ^{234}U critical level to the ^{238}U result is less than 1.
- If ^{234}U is detected, the uranium should be considered from an occupational source if the ratio of the ^{234}U result to the ^{238}U critical level is greater than 3.

In vitro analysis for radiostrontium

The start date for urinalysis for radioisotopes of strontium was not specifically determined, other than "late 1950s." From start until 1969 strontium was separated by alkaline earth phosphate coprecipitation followed by beta counting on a GM counter. This analysis was also called fission product analysis (see discussion below). Both ^{89}Sr and ^{90}Sr would have been counted (and potentially radioisotopes of cerium and promethium), but it is claimant-favorable to assume the result is all ^{90}Sr. Yttrium-91, grown in from ^{91}Sr, might also be present at intake if the source of intake was recent fission products (up to about a year post irradiation). If the intake prior to 1969 occurred in reactor buildings or the 221-F canyon facility and an intake of ^{90}Sr is indicated, then an equal intake of ^{91}Y should be assumed. This is a claimant-favorable assumption because after about a year post irradiation, ^{91}Y activity would be negligible. The reporting level is assumed to be the same as for the beta component of the fission product analyses.

From 1969 to 2001 strontium was analyzed by liquid ion-exchange that separates the yttrium progeny, followed by beta proportional counting. Yttrium-91 would be included as a possible interference but ^{89}Sr would not. For ^{90}Sr intakes in 1969 to 1989 in reactor buildings, an intake of ^{89}Sr equal to the intake of ^{90}Sr should be assumed.

Since 2001, ^{90}Sr has been separated as part of the use of TRU and TEVA resins for plutonium, uranium, actinides, and strontium.

In vitro fission product analysis

Prior to whole body counts, intakes of gamma-emitting fission products were monitored by urinalysis. The record is not clear exactly what procedure was used; both gamma-spectrometry analyses and gross beta counting are mentioned as being possible. Nor has it been determined if any separation chemistry was done prior to counting, though the reporting levels that were used seem to imply that ^{40}K was removed or reduced prior to counting. These analyses were coded FP, FPIA, or IA under element on the employee bioassay cards. The IA code can have two meanings depending on the context. An IA code with a result prior to 1960 most likely refers to the fission product analysis; an IA code with no result means inconclusive analysis, that is, the lab was unable to produce a valid result. The employee bioassay cards show less-than values that varied over the years, presumably indicating different reporting levels and probably indicating a switch from beta counting to photon counting; for instance, 1955-1960: <30 d/m/750 ml; 1960-65: <100 d/m/1.5 L; 1966-1971; <1 nCi/1.5L. Some records in the 1960-65 period show two results for the same sample: one for beta (<100 dpm/1.5L) and one for gamma (<500 dpm/1.5L). This would imply they did a gamma scan, probably on the whole sample before the chemistry. The records also show some fission product urinalyses as late as 1984 with the result listed as <1. Presumably the units are nCi/1.5L.

A procedure for fission product urinalysis at Hanford, dated May 1950, has been discovered (Thorburn, 1950). It is likely that the Savannah River procedure was similar; at least through 1960. The procedure separated and counted radionuclides of strontium, yttrium, cerium, and promethium. It did not account for radionuclides of ruthenium or cesium.

To interpret results from the fission product urinalysis, it is noted that the strontium, barium, europium, zirconium, and niobium radionuclides concentrate primarily in the bone, with ^{90}Sr providing the largest dose rate. Cerium, lanthanum, and promethium concentrate primarily in the liver, with some concentration in the bone, with ^{144}Ce providing the largest dose rate. Cesium and ruthenium are assumed to be uniformly distributed in the whole body.

Hence, based on what little is known, fission product urinalysis results should be interpreted as follows:

Note: If the claimant had a whole body count during any year and had a detectable fission product urinalysis result, the intakes of ^{137}Cs, ^{106}Ru, or any other gamma-emitting fission products should be determined from the whole body count.

- 100% Type F ^{90}Sr if the claimant's disease is related to bone or bone marrow plus an equal intake of ^{137}Cs (Type F) and 3 fold intake of ^{106}Ru (Type F).
- 100% Type W ^{144}Ce if the claimant's disease if related to the liver plus an equal intake of ^{106}Ru and a 20% intake of ^{137}Cs (Type F).
- For any other disease location, the fission product analysis should be assumed Type F ^{90}Sr to be an indicator of a ^{137}Cs intake (although it probably did not measure ^{137}Cs directly). Hence, calculate an intake of ^{90}Sr from the fission product results and add an equal intake of ^{137}Cs and 3 fold intake of ^{106}Ru.

In vitro analysis for ^{237}Np

The Savannah River Technical Basis Manual, Rev. 8, says that neptunium analyses started in the 1950s, perhaps 1959, using coprecipitation with calcium-magnesium ammonium phosphate and ion exchange, followed by autoradiography on NTA emulsion for 10,000 minutes. The MDA was not given but was mostly likely similar to the MDA for plutonium alpha autoradiography method used at the same time, i.e., about 0.035 dpm/L. The results were recorded as S.P. (special product), and the reporting level was 0.05 dpm/1.5 L, same as plutonium. Analysis by TIOA extraction and electrodeposition occurred at the same time as for plutonium analyses, about 1964, which was a faster method but the reporting level stayed the same. In 1993-1996, samples were analyzed by an offsite, commercial laboratory using extraction chromatography resin. In 1996 analysis returned to SRS and alpha spectrometry was introduced. Since 2001, TEVA and TRU resins are used to separate neptunium from actinides and uranium, and alpha spectrometry has continued. The MDA is still listed as 0.035 dpm/L; however, from personal communication with the SRS lead internal dosimetrist, an MDA of 0.4 dpm/L was considered more applicable in practice. If results are reported below 0.4 dpm/L, they can be used as given.

4.1.3 Correcting for Urinalysis Volume

At SRS, routine urine samples are considered partial-day samples. Urinalysis results are recorded as per 1.5L or per L. The code used to determine intake from bioassay data requires the period of time represented by the sample and the result of the sample in activity per day. If the volume of the sample is known, the result can be adjusted to activity per sample. Then if the collection time of the sample is known, the activity can be adjusted to activity per day by ratioing the collection hours to 24 hours. (This is often

the case for special samples collected since about 1990.) If either the collection period or the volume is not known, then a result in activity per L or per 1.5 L should be normalized to activity per day using the reference volume of 1.4 L per day.

4.1.4 Fecal Sample Analysis

SRS technical documentation indicates that fecal sample analysis has been performed periodically since the 1950s. These samples were always for special, rather than routine, sampling purposes, and were always performed in conjunction with urine bioassay sampling. Fecal results will be found on separate reports, not routine employee bioassay cards.

Fecal samples are screened by photon counting with the following typical MDAs: 7 pCi (^{241}Am), 300 pCi (^{238}Pu), and 600 pCi (weapons grade plutonium). However, if results could not be quantified by photon counting, that is, if nothing was detected, then samples may have been analyzed by normal separation chemistry and alpha spectrometry to improve the detection capability. This protocol applies mostly to samples from 1994 to present, and probably was not applied to samples prior to 1994. Because only an aliquot of the fecal sample is used for wet radiochemistry, the MDAs are typically 1.5 times the MDAs for routine urine samples.

4.2 *In vivo* MDAs, Counting Methods, and Reporting Practices

Whole body counting has been performed since 1960 when the SRS Whole Body Counting facility was completed. Chest counting was initiated 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 hard-copy record. A whole body count record and a chest count record on the same day are listed on a single hard-copy record. *In vivo* bioassay methods and detection capabilities are reviewed below.

4.2.1 Whole Body Counting

Prior to whole body counting, fission products intakes were monitored using urinalysis.

Whole body counting began in approximately 1960, using a 4” high by 8” thick diameter NaI detector. (Detection energy range was 100 keV to 2000 keV.) The monitored individual sat in a reclining chair positioned in an arc around the detector; this was referred to as the “40-cm arc geometry” in bioassay monitoring reports. Bioassay via this method was not used for plutonium and americium due to their low energy emissions. It is reported this detector was in service until September 1995. Reported MDAs for various radionuclides are shown in Tables 4.2.1-1 and 4.2.1-2. MDAs and reporting levels were the same

Table 4.2.1-1 Whole Body Counting

Period	Nuclide	MDA, nCi	MDA method
~1960 - ?	Ce-144	29	~3X standard deviation of expected count rate in energy region
	U[(a)]	62	
	I-131	1.4	
	Ru-106	6.1	
	Cs-137	1.0	
	Zr/Nb-95	2.2	
	Zn-65	5.1	
	Ba/La-140	9.3	

(a) Listed as U, but based on measurement of U-235. A result greater than the MDA would certainly be a false positive unless it was associated with a major intake. Other counts and urinalyses would be expected.

In the early 1970s, whole body counting using an array of NaI detectors was implemented. (WSRC-IM-90-139 indicates five 4"x4" detectors were used, while WSRC-RP-95-234 says that four 4"x5" detectors were used.) Additionally, in the mid-1980s, whole body counting using stand-up geometry and large (4"x4"x16" or 5"x3"x16") NaI detectors was implemented. MDAs for these counting systems were generated individually for counts by processing software. Many forms for reporting the results of whole body counts were used throughout the years. Some had the MDAs listed on the form; others just reported "detected" radionuclides. If MDAs are not shown, use the ones in Table 4.2.1-1.

Note: Some of the forms use the abbreviation nc for nanocurie.

Table 4.2.1-2 Current MDAs for Whole Body Counting

Radionuclide	MDA (nCi)	Radionuclide	MDA (nCi)	Radionuclide	MDA (nCi)
Mn-54	3.4	Sb-125	14	Eu-152	18
Co-60	2.9	Cs-134	3.8	Eu-154	8.4
Zn-65	6.1	Cs-137	4.1	U-235	14
Ru-106	36	Ce-144	69	Np-237	14

For special investigational counts, the MDAs are stated to be approximately 0.71 times these values for whole body counting.

Many SRS workers had/have burdens of ^{137}Cs from non-occupational sources, e.g., fallout and consumption of venison. The SRS presently uses a value of 20 nCi of ^{137}Cs in a whole body count for consumers of venison as the decision level to follow up or not. If the record clearly indicates that the worker was a consumer of meat from wild game harvested on the mid-to-southern Atlantic seaboard extending into the Appalachian Mountains, ^{137}Cs results in whole body counts can be disregarded unless there is also an indication of intake of other fission/activation products or ^{90}Sr in a urine sample. Otherwise, it is claimant-favorable to assume the ^{137}Cs resulted from an occupational intake.

4.2.2 Chest Counting

Chest counting was initially performed using a 5" diameter thin NaI detector from 1966 to around 1971. No MDA information could be located in SRS technical documentation for this method of analysis.

In 1971, a phoswich detection system was implemented that used a NaI/CsI sandwiched detector. This system was modified to a dual phoswich system in 1972, which remained in use until around 1987 or 1988. MDAs were listed on results forms that varied from count to count but were in the neighborhood of 0.2 nCi for ^{241}Am, 5 nCi for ^{238}Pu, 10 nCi for ^{239}Pu, and 5 nCi for ^{244}Cm; however, based on operating experience over the years, the values for all but the ^{241}Am seem overly optimistic due to noise in the 1-20 keV part of the spectrum. The method of accounting for chest wall thickness may also be a factor in the difference between stated MDAs and present experience. However, phoswich systems had better low-energy background and better detection efficiency than germanium systems. Phoswich systems had poor resolution and could not discriminate between ^{238}Pu, ^{239}Pu, or ^{244}Cm.

In general chest counting MDAs should not be used to determine potential undetected intakes of plutonium if urinalysis data are available.

The current six detector solid state planar germanium array was implemented in 1987 or 1988.

MDAs for later periods have been generated individually for counts by processing software, with typical values shown in Table 4.2.2-1, based on a chest wall thickness of 2.5 cm. However, MDAs are often listed for each count as part of the measurement results and the count-specific MDAs should be used when available. The uncertainty in the MDA calculation has been estimated, according to WSRC-IM-90-139, rev. 8, as ±46%. The confidence level was not stated but would appear to be one standard deviation. However, uncertainty in chest wall thickness can have a large impact on MDA determination. Using weight and height measurements on 1100 male workers at Hanford in 2002-2003, and using the chest wall thickness formula in the SRS technical basis manual, the 50th percentile chest wall thickness was 3.7 cm and the 95th percentile chest wall thickness was 5.9 cm. The MDAs for radionuclides with low-energy photons affected by variations in chest wall thickness for the 50th and 95th percentile male workers are shown in Table 4.2.2-2.

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Table 4.2.2-1 Current MDAs for Chest Counting (2.5 cm chest wall thickness)

Radionuclide	MDA (nCi)	Radionuclide	MDA (nCi)	Radionuclide	MDA (nCi)
Ce-144	0.31	U-235	0.10	6% Pu	96
Eu-152	0.056	U-236	89	12% Pu	70
Th-228	3.2	U-238	1.1	Am-241	0.10
Th-232	31	Np-237	0.31	Am-243	0.12
DU	1.2	Pu-238	58	Cm-242	28
RU	8.3	Pu-239	130	Cm-244	37
HEU	5.2	Pu-240	47	Cf-252	32
U-234	30	3% Pu	110		

Table 4.2.2-2 Current MDAs for Chest Counting 50th Percentile and 95th Percentile Male Workers

Radionuclide	MDA (nCi) 50th	MDA (nCi) 95th
Am-241	0.13	0.20
Cm-244	78	310
Pu-238	130	580
Pu-239	290	1300
Pu-240	110	470
Cf-252	85	500

For special investigational chest counts, the MDAs are stated to be approximately 0.77 times these values.

For ^{241}Am results associated with an intake of 6% to 12% ^{240}Pu mixture, the dose reconstructor can assume that the plutonium isotopes behave the same in the lung as the ^{241}Am. This assumption does not hold for systemic americium, however.

4.3 Personal Air Sampling Data as Applicable

The objective of breathing zone air monitoring is to measure the concentration of radioactive material in the air that a worker breathes for the purpose of calculating exposure [in Derived Air Concentration-hours (DAC-h)]. Intakes calculated from air monitoring were assigned only if bioassay data are unavailable or inadequate. Breathing zone air monitoring must sample air that is representative of the air in the worker's breathing zone. This was accomplished by placing the sampler as close as possible to the worker's nose and mouth. Fixed breathing zone air samplers are typically placed at face height in front of hoods and glove boxes. Portable breathing zone air samplers, often referred to as personal air samplers (PAS), are attached to the worker's clothing in the neck region. PAS have been used when it was not practical to use regional air samplers (RAS) (such as work outside of buildings), inside plastic suits used in highly contaminated environments, and for operational studies.

Air sample records were considered workplace monitoring records as opposed to intake monitoring records and 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. Boxes containing workplace monitoring records can be retrieved from storage at the federal repository; however, locating a specific set of air sample records would be very labor consuming and should be considered only as a last resort. Correlation between air sample concentrations in given rooms or work locations and a specific person would be difficult.

4.4 Interferences, Uncertainty, and Use of Reporting Levels

4.4.1 Contamination of Samples

Excreta samples were collected in the workplace. Because the levels of activity that are significant in excreta samples, especially urine samples, were generally below detectability on workplace personnel detectors, contamination of samples from the worker’s hands or clothing is a possibility. Hanford found a decrease in detectable

plutonium bioassay results after switching to home collection. Laboratory contamination and mix-up of samples in the laboratory are also a possibility, although laboratory Quality Control procedures and performance of test samples was designed to minimize this source of contamination.

It is likely that a contaminated sample will show up as an obvious outlier in the dataset for a given worker. If the dataset shows an unusually high urinalysis result for a radionuclide other than tritium, and if follow-up samples were collected that were not consistent with the high result, then the high result may be considered an outlier. However, if the result is not obviously an outlier, then it is claimant-favorable to assume the result is real.

For in vivo measurements, contamination can occur as external to the body or, in the case of chest counting, as external to the lung. If a follow-up in vivo count is obtained the same day or within a few days that shows a dramatic decrease in activity or no detectable activity, then external contamination can be assumed. Radon progeny and medical diagnostic or therapeutic procedures involving radionuclides can cause interferences to in vivo measurements, especially for NaI detectors. However, unless the count was invalidated or noted as being influenced by such interferences, the results should be used as recorded.

4.4.2 Uncertainty

Uncertainties for the bioassay measurements were not stated in the records. For results near or at the reporting levels, the assumption provided in the *Internal Dose Reconstruction Implementation Guide* (NIOSH 2002b) should be used, namely the standard deviation is 0.3 times the MDA or reporting level. For results greater than 3 times the MDA or reporting level, the standard deviation can be assumed to be 0.1 times the result, based on Currie's quantification level (Currie 1968). If actual standard deviations or other indications of error are reported with a bioassay measurement result, the reported value should be used.

4.4.3 Use of Reporting Levels and Less-Than Values

<u>Special consideration for plutonium intakes</u>

Because plutonium stays in the body for a very long time, and because the urine excreta curve (activity per day excreta versus days after intake) has a small slope beyond the first year, an intake of plutonium that might have been missed in the 1950s or 60s because of poor detection capability or other reasons can still be confirmed or otherwise by urinalysis obtained years later. Additionally, low-level chronic or frequent-intermittent intakes not originally detectable can, if continued for many years, will lead to increasing rates of excreta over time. This condition has implications for the determination of potential missed intakes.

Table 4.4.3-1 provides the maximum potentially missed chronic intakes of ^{239}Pu based on a urinalysis MDA of 0.1 dpm/day (0.07 dpm/L) for various periods of chronic intake and two commonly used particle size distributions. The urine sample is obtained on the last day of the period (under Duration of Intake). This approach provides the maximum

potentially missed chronic intake regardless if there were missed samples or samples with high MDAs during the period. Once the ^{239}Pu intake is obtained, standard isotopic ratios for (e.g., 6% ^{240}Pu or 12% ^{240}Pu) can be applied to get the total intakes.

Table 4.4.3-1 ^{239}Pu Chronic Inhalation Intake Assessment Based on MDA Urinalysis on Last Day of the Period

Solubility:		**Type M**			**Type S**		
Particle Size:		5			5		
Analytical MDA:		0.1			0.1		
Duration of Intake		**Daily Intake Rate**	**Cumulative Intake**		**Daily Intake Rate**	**Cumulative Intake**	
(years)	**(days)**	**(dpm/d)**	**(dpm)**	**(nCi)**	**(dpm/d)**	**(dpm)**	**(nCi)**
5	1825	15.86	2.89E+04	13.0	293.4	5.35E+05	241
10	3650	11.37	4.15E+04	18.7	164.3	6.00E+05	270
15	5475	9.432	5.16E+04	23.3	121.6	6.66E+05	300
20	7300	8.243	6.02E+04	27.1	99.17	7.24E+05	326
25	9125	7.410	6.76E+04	30.5	84.98	7.75E+05	349
30	10950	6.785	7.43E+04	33.5	75.09	8.22E+05	370
35	12775	6.293	8.04E+04	36.2	67.76	8.66E+05	390
40	14600	5.894	8.61E+04	38.8	62.11	9.07E+05	408
45	16425	5.564	9.14E+04	41.2	57.59	9.46E+05	426
50	18250	5.284	9.64E+04	43.4	53.90	9.84E+05	443

Solubility:		**Type M**			**Type S**		
Particle Size:		1			1		
Analytical MDA:		0.1			0.1		
Duration of Intake		**Daily Intake Rate**	**Cumulative Intake**		**Daily Intake Rate**	**Cumulative Intake**	
(years)	**(days)**	**(dpm/d)**	**(dpm)**	**(nCi)**	**(dpm/d)**	**(dpm)**	**(nCi)**
5	1825	10.00	2.01E+04	9.04	164.2	3.00E+05	135
10	3650	7.847	2.86E+04	12.9	90.0	3.29E+05	148
15	5475	6.500	3.56E+04	16.0	66.24	3.63E+05	163
20	7300	5.677	4.14E+04	18.7	53.90	3.94E+05	177
25	9125	5.102	4.66E+04	21.0	46.14	4.21E+05	190
30	10950	4.670	5.11E+04	23.0	40.75	4.46E+05	201
35	12775	4.331	5.53E+04	24.9	36.77	4.70E+05	212
40	14600	4.056	5.92E+04	26.7	33.70	4.92E+05	222
45	16425	3.828	6.29E+04	28.3	31.26	5.13E+05	231
50	18250	3.635	6.63E+04	29.9	29.26	5.34E+05	241

5.0 OCCUPATIONAL EXTERNAL DOSIMETRY

5.1 Introduction

SRS operations involved several processes of the nuclear weapons development cycle (DOE 1997) and represented a significant role within the US nuclear weapons program. These processes include nuclear fuel fabrication; nuclear reactor operations; radiochemical separations; radionuclide production; recycling uranium; refining, finishing and storing plutonium; and handling the associated radioactive waste.

SRS workers, especially those employed during the peak production decades of the 1950s and 1960s, have been exposed to radiation types and energies associated with the respective nuclear weapon development processes. SRS utilized facility and individual worker monitoring methods to measure and control radiation exposure to workers. There are numerous SRS records concerning facility monitoring, safety evaluations, investigations, etc. However, it is time consuming to locate and evaluate these records for SRS facilities and processes since 1951. Evaluations of the extensive scope of facility, process, and worker information relevant to an individual worker's potential dose, many years or even decades after employment are difficult. Records of radiation dose to individual workers from personnel dosimeters worn by the worker and co-workers are available for SRS operations beginning in 1951. Dose from these dosimeters was recorded at the time of measurement, routinely reviewed by SRS operations and radiation safety staff for compliance with radiation control limits, and routinely available to individual workers. The NIOSH External Dosimetry Implementation Guide (NIOSH 2002a) has identified these records to represent the highest quality record to retrospectively assess dose. Information in this section pertains to analyzing these records and does not address parameters regarding skin and testicular and/or breast radiation dose that may result from acute beta (electron) radiation exposure in generally non-routine workplace exposure profiles.

Radiation dosimetry practices were initially based on experience gained during several decades of radium and x-ray medical diagnostic and therapy applications. These methods were generally well advanced at the start of the Manhattan Engineering District (MED) program to develop nuclear weapons beginning about 1940. The primary difficulties encountered by MED operations to measure worker dose to external radiation involved:

- Large quantities of high level radioactivity
- Mixed radiation fields involving beta, photon (gamma and x-ray), and neutron radiation with low, intermediate, and high energies.
- Neutron radiation.

5.2 Basis of Comparison

Historically, since the initiation of the MED in the early 1940s, various radiation dose concepts and quantities have been used to measure and record occupational dose. The basis of comparison for reconstruction of dose, as described in "External Dose Reconstruction Implementation Guideline" (NIOSH 2002a), is the Personal Dose Equivalent, Hp(d), where d identifies the depth (in mm) and represents the point of reference for dose in tissue. For weakly

penetrating radiation of significance to skin dose, d = 0.07 mm and is noted as Hp(0.07). For penetrating radiation of significance to "whole body" dose, d = 10 mm and is noted as Hp(10). Both Hp(0.07) and Hp(10) are the radiation quantities recommended for use as the operational quantity to be recorded for radiological protection purposes by the International Commission on Radiological Units and Measurements (ICRU 1993). In addition, Hp(0.07) and Hp(10) are the radiation quantities used in the DOE Laboratory Accreditation Program (DOELAP) used to accredit DOE personnel dosimetry systems since the 1980s (DOE 1986). The International Agency for Research on Cancer (IARC) Three Country Combined Study (Fix et al 1997) and IARC Collaborative Study (Thierry-Chef et al 2002) selected Hp(10) as the quantity to assess error in historical recorded "whole body" dose for workers in the IARC nuclear worker epidemiologic studies. The basis for comparison for neutron radiation is more complicated since historically the calibration of dosimeters to measure neutron dose was based on different dose quantities such as First Collision Dose, Multiple Collision Dose, Dose Equivalent Index, etc. The numerical difference in using these dose quantities compared to the Hp(10) dose used in current DOELAP performance testing could be evaluated using the relative values of the dose conversion factors for the respective dose quantities in conjunction with characteristics of the respective SRS neutron dosimeter response characteristics and workplace radiation fields.

5.3 Dose Reconstruction Parameters

Examinations of the beta, photon (x-ray, gamma ray), and neutron radiation types energies and geometries of exposure, and the characteristics of the respective SRS dosimeter response are crucial to the assessment of bias and uncertainty of the original recorded dose in relation to the radiation quantity Hp(10). The bias and uncertainty for current dosimetry systems is typically well documented for Hp(10) (SRS 1993). Often the performance of current dosimeters can be compared with performance characteristics of historical dosimetry systems in the same, or highly similar, facilities or workplaces. In addition, current performance testing techniques can be applied to earlier dosimetry systems to achieve a consistent evaluation of all dosimetry systems. Dosimeter response characteristics for radiation types and energies in the workplace are crucial to the overall analysis of error in recorded dose.

Overall, accuracy and precision of the original recorded individual worker doses and their comparability to be considered in using NIOSH (2002) guidelines depend on (Fix et al 1997):

- **Administrative practices** adopted by facilities to calculate and record personnel dose based on technical, administrative, and statutory compliance considerations.

- **Dosimetry technology** includes the physical capabilities of the dosimetry system, such as the response to different types and energies of radiation, in particular in mixed radiation fields.

- **Calibration** of the respective monitoring systems and similarity of the methods of calibration to sources of exposure in the workplace.

- **Workplace radiation fields** that may include mixed types of radiation, variations in exposure geometries, environmental conditions.

An evaluation of the original recorded doses based on these parameters is expected to provide the best estimate of Hp(0.07), as necessary, and Hp(10) for individual workers with the least relative overall uncertainty.

5.3.1 SRS Historical Administrative Practices

Historically, SRS had an extensive radiation safety monitoring program to measure radiation exposure in the workplace using portable radiation instruments, contamination surveys, zone controls, and personnel dosimeters. The results from the personnel dosimeters were used to measure and record dose from external radiation exposure to SRS workers throughout the history of SRS operations (Taylor et al 1995). These dosimeters include one or more of the following:

- Personnel Whole Body (WB) beta/photon dosimeters
- Pocket Ionization Chamber (PIC) dosimeters
- Personnel extremity dosimeters
- Personnel neutron dosimeters.

SRS began operations in 1951 using dosimeter and processing technical support provided by the Oak Ridge National Laboratory (ORNL). A summary of SRS personnel beta/photon and neutron (respectively, dosimeter type, exchange, Minimum Detection Level (MDL) and potential missed annual dose) is shown in Table 5.3.1-1. ORNL, then the Clinton Laboratory, had implemented their dosimetry methods based on the personnel beta/photon dosimeter design developed at the Metallurgical Laboratory at the University of Chicago (Pardue et al 1944). ORNL provided SRS with beta/photon film dosimeters and neutron nuclear track, type A (NTA) emulsion dosimeters at the beginning of SRS operations until SRS implemented their in-house personnel dosimetry capabilities about one year later. The precise detection levels shown in Table 5.3.1-1 are difficult to estimate, particularly for the older systems. The current SRS commercial TLD system minimum detection levels (MDLs) are identified in SRS external dosimetry technical basis documentation (SRS 1993) based on a DOE Laboratory Accreditation Program (DOELAP) laboratory testing protocol (DOE 1986). During earlier years, the MDLs are subject to additional uncertainty since factors involving the radiation field and film type, as well as the processing, developing and reading system cannot be tested. The film dosimeter MDLs shown in Table 5.3.1-1 were estimated based on information from NIOSH (1993), NRC (1989), and Wilson et al (1990). The TLD MDLs were obtained from Taylor et al (1995) and SRS (1993).

Dose reconstruction parameters concerning SRS administrative practices significant to dose reconstruction involve issues as follows:

- Policies to assign dosimeters to workers
- Policies to exchange dosimeters
- Policies to record notional dose (i.e., some identified value for lower dosed workers often based on a small fraction of the regulatory limit)
- Policies to estimate dose for missing or damaged dosimeters
- Policies to replace any destroyed or missing records
- Policies to evaluate and record dose for incidents

- Policies to obtain and record occupational dose to workers for other employer exposure.

Table 5.3.1-1 SRS Dosimeter Type, Period of Use, Exchange Frequency, MDL and Potential Annual Missed Dose

Dosimeter	Period of Use	Exchange Frequency	Laboratory MDL (mSv)[(a)]	Max. Annual Missed Dose (mSv)[(b)]
Beta/Photon Dosimeters				
ORNL Two-Element film Dosimeter	Prior to 3/52	Weekly (n = 52)	0.4	20.8
SRS Two Element film Dosimeter	After 3/52 through 9/30/1957	Weekly (n = 52)	0.4	20.8
	Beginning 10/1/1957 through 11/8/1959	Biweekly (n=26)	0.4	10.4
SRS Multi-Element Film Dosimeter	Beginning 11/9/1959 through 12/31/1964	Biweekly (n=26)	0.4	10.4
	Beginning 1/1/1965 through 12/31/1965	4-week (n=13)	0.4	5.2
	Beginning 1/1/1966 through 3/31/1970	Monthly (n=12)	0.4	4.8
SRS TLD	Beginning 4/1/1970 through 6/30/1983	Monthly (n=12)	0.15	1.8
Panasonic TLD	Beginning 7/1/1983 to 2003 (ongoing)	Monthly (n=12)	0.05	0.6
Neutron Dosimeters				
ORNL NTA	Prior to 8/3/1953	Weekly (n = 52)	~0.5	13[(c)]
SRS NTA	Beginning 8/3/1953 through 7/13/1960	Weekly (n = 52)	~0.5	13[(c)]
SRS NTA	Beginning 7/14/1960 through 12/31/1970	Biweekly (n=26)	~0.5	6.5[(c)]
SRS Hoy TLND	Beginning 1/1/1971 through 12/31/1994	Monthly (n=12)	0.2	1.2
Panasonic TLND	Beginning 1/1/1995 to 2003 (ongoing)	Monthly (n=12)	0.15	0.9

a. Estimated film dosimeter detection levels based on NIOSH (1993), NRC (1989), and Wilson et al (1990). TLD detection levels from Taylor et al (1995) and SRS (1993).
b. Maximum annual missed dose (NIOSH 2002)
c. The potential annual missed dose based on laboratory irradiations is not applicable to workplace missed neutron dose.

SRS policies appear to have been in place for all of these parameters. SRS routine practices appear to have required assigning dosimeters to all workers who entered a controlled radiation area. Dosimeters were exchanged on a routine schedule. All dosimeters were processed and the measured results recorded and used to estimate dose. There appears to be no use of recorded notional doses, although there are issues of recording dose for low-dose dosimeters (See section on "missed dose"). Figure 5.3.1-1 illustrates the number of SRS monitored workers and the collective "whole body" dose of record for SRS workers since 1951 (Taylor et al 1995). Administrative practices are generally available in the SRS procedures manual (DuPont 1954, 1957, 1961a, 1961b), and detailed information for each worker in the respective SRS exposure history file documentation. Based on the SRS documentation and the reasonable trend in these

figures, there does not appear to be any significant administrative practice that would affect the integrity of the recorded dose of record for SRS workers with the exception that SRS hard-copy radiation records must be consulted to separate the photon, neutron and tritium dose for workers who terminated prior to 1979. Apparently, only the total whole body dose (i.e., photon + neutron + tritium) is available from the second quarter of 1963 through December 31, 1972. The missing dose components of the total whole body dose could be estimated from prior year and subsequent year dose results according to the methods described by Watson et al (1994) based on examination of continuity in the worker's job and work activities.

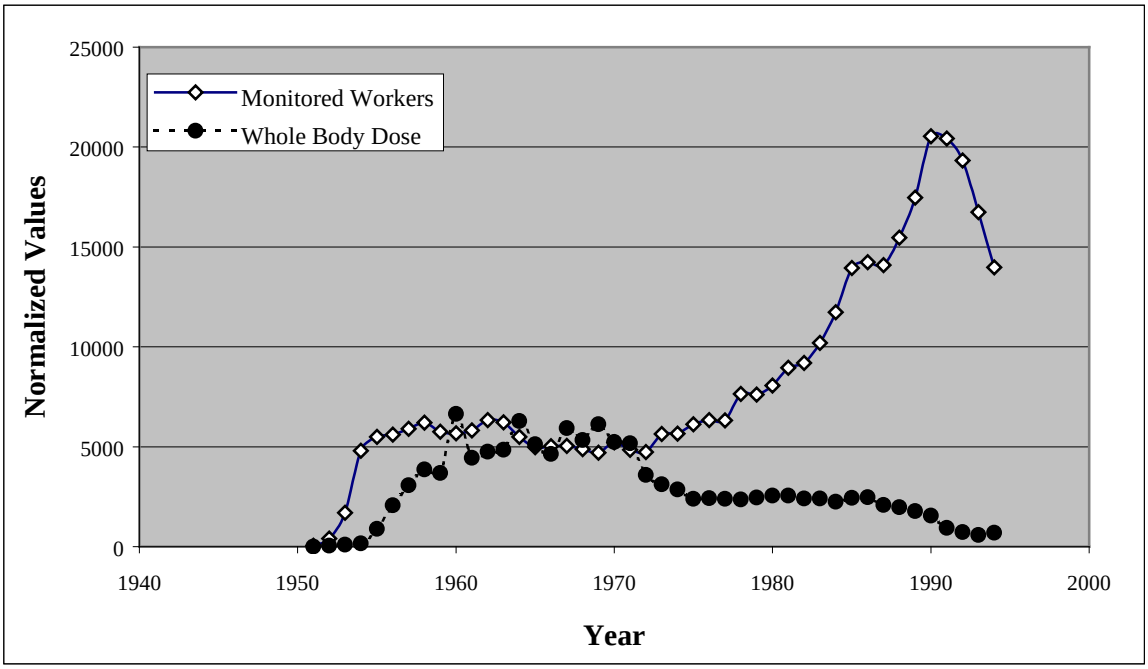

Figure 5.3.1-1 SRS Monitored Workers and Collective Dose

5.3.2 SRS Dosimetry Technology

SRS dosimetry methods evolved during the years as improved technology was developed and the complex radiation fields were better understood. The adequacy of the respective dosimetry methods to accurately measure radiation dose is determined from the radiation type, energy, exposure geometry, etc., as described in later sections. The dosimeter exchange frequency was gradually lengthened, generally corresponding to the time period of the regulatory dose controls. At the beginning of SRS operations, a weekly dose control of 3 mSv was in effect. This was changed to an annual limit of 50 mSv in the latter 1950s. Table 5.3.2-1 is a summary of the major historical events in the SRS personnel dosimetry program.

Operations at SRS occurred about 7 to 8 years after the MED nuclear weapons facilities began operations at Oak Ridge National Laboratory (ORNL), Los Alamos National Laboratory (LANL), and Hanford. These AEC sites implemented, in the beginning, essentially the same beta/photon and neutron dosimeter methods. There were

intercomparisons of dose interpretations among these AEC sites through the years as reported in Wilson et al (1990).

Table 5.3.2-1 SRS Historical Dosimetry Events (Taylor et al 1995)

Date	Description
1951	Pocket Ionization Chambers (PICs) were used to measure SRS worker exposure. PICs were used in addition to film dosimeters.
11/12/1951	ORNL two-element (i.e., open window and 1 mm silver filter) beta/photon dosimeters issued to SRS workers. Processing was provided by ORNL. Routine dosimeter exchange period was weekly.
1951	Nuclear Track Emulsion (NTA) dosimeters were issued to selected SRS workers to measure neutron radiation. Capability was provided by ORNL. NTA film was placed, facing the worker's body, in two-element beta/photon dosimeter holders. NTA film exchange was also weekly.
3/3/1952	SRS began onsite dosimetry program with the same two-element film dosimeter design as ORNL.
3/1953	SRS began processing film using the ORNL film badge dosimeter. The results were reported in mrem/week, manually recorded on individual worker file folders.
1953	Extremity film dosimeter use and processing began at SRS.
8/3/1953	SRS assumed processing of NTA films.
4/1954	Indium foil placed in dosimeter holder to respond to neutron radiation.
10/1/57	SRS beta/photon dosimeter exchange changed to biweekly.
11/9/1959	SRS designed multi-element film dosimeter use was implemented. This dosimeter design permitted analysis of beta, gamma and x-ray exposure to personnel.
7/14/1960	NTA exchange changed to biweekly.
1963	PICs were replaced with self-reading dosimeters to provide a real-time indication of exposure.
1965	Four week beta/photon dosimeter exchange implemented.
1966	Monthly beta/photon dosimeter exchange implemented.
4/1/1970	SRS TLD replaced film as the principal means of measuring beta/photon dose. The minimum recorded dose was about 0.15 mSv.
1/1/1971	SRS Hoy Thermoluminescent Neutron Dosimeter (TLND) implemented.
1980	Extremity dose routinely measured with TLD chip.
7/1/1982	Commercial Panasonic beta/photon TLDs implemented.
mid-1980s	SRS implemented changes such as on-phantom calibration and 137Cs calibration source to participate in new DOE Laboratory Accreditation Program (DOELAP) implemented to performance testing. The overall change in recorded dose with new practices, as described in Taylor et al (1995), was determined to be an equivalent 11.9% increase compared to earlier dose estimates.
1989	SRS TLD is DOELAP accredited. Lower limit of detection, based on DOELAP protocol, was 0.2 and 0.05 mSv, respectively, for the shallow and deep dose components.
1/1/1995	Panasonic neutron dosimeter replaced SRS TLND. Routine dosimeter exchange for the Panasonic beta/photon dosimeter is quarterly and for the Panasonic TLND, monthly.

5.3.2.1 *Beta/Photon Dosimeters*

In the early 1950s, SRS used personnel dosimeter services from ORNL (Taylor et al 1995). ORNL had implemented in the mid-1940s the beta/photon film dosimeter design, with perhaps minor differences, that was developed originally at the Metallurgical Laboratory at the University of Chicago (Pardue et al 1944). Within about one year, SRS implemented their own in-house dosimetry capabilities. SRS followed a process of dosimetry research and development leading to gradual upgrades in dosimetry capabilities for the complex radiation fields encountered in SRS reactor, reprocessing and plutonium handling facilities (Taylor et al 1995). This evolution in dosimetry capabilities was followed by other DOE sites leading to site-specific multi-element film and thermoluminescent dosimetry systems. SRS dosimeters always included a comparatively "open window" to measure significant beta radiation and to distinguish the dosimeter response between beta and photon radiation. The photon energy response characteristics of the SRS beta/photon dosimeters are illustrated in Figure 5.3.2.1-1 based on the essentially identical two-element film dosimeter used at SRS, Hanford and ORNL (and likely other MED sites) designed at the University of Chicago (Pardue et al 1944). Figure 5.3.2.1-1 also illustrates the Hp(10) response. The improved energy response of multi-element film and thermoluminescent dosimeters is shown in Figure 5.3.2.1-1 using available Hanford published information (Wilson et al 1990). The general trend in the energy response curves for these dosimeters, relative to Hp(10), provides insight into the expected performance of these dosimeters in SRS workplace radiation fields. The original two-element dosimeter shows a significant under-response to photon energies less-than about 100 keV. Evaluations of the two-element film dosimeter that was used at SRS, ORNL, LANL, and Hanford sites have been performed by the International Agency for Research on Cancer (IARC) Collaborative Study researchers (Thierry-Chef et al 2002) and by Fix et al (1994) to several photon energies. IARC conducted a dosimeter intercomparison study to higher-energy (i.e., >100 keV) photons of ten commonly used historical dosimetry systems throughout the world. Three of the dosimeter designs were from the US. These included the two-element dosimeter used at Hanford (identified as US-2), multi-element film dosimeter used at Hanford (identified as US-8), and the currently used SRS Panasonic dosimeter system (identified as US-22). The IARC Study considered that exposure to dosimeters worn by workers could be characterized as a combination of anterior-posterior (A-P), rotational, and isotropic irradiation geometries. Dosimeter response to selected photon energies was measured using two phantoms. These phantoms were used to simulate the effect of the workers body on the measured dosimeter response. The first phantom is the International Standards Organization water filled slab phantom with polymethyl methacrylate walls measuring 30 cm wide by 30 cm tall and 15 cm in depth that is widely used for dosimeter calibration and performance testing. The second phantom used was an anthropomorphic Alderson Rando Phantom. This phantom is constructed from a natural human skeleton cast inside material that has a tissue equivalent response. The results of this testing, for US dosimeters only, are presented in Table 5.3.2.1-1. As noted in Table 5.3.2.1-1, the two-element dosimeter design significantly overestimated Hp(10) for all irradiation geometries and increasingly for the lower energy photon energy at 118 keV. Similar anthropomorphic phantom laboratory irradiations using anterior-posterior and rotational geometries to several photon energies have been done at Hanford (Wilson et al 1990) with similar results.

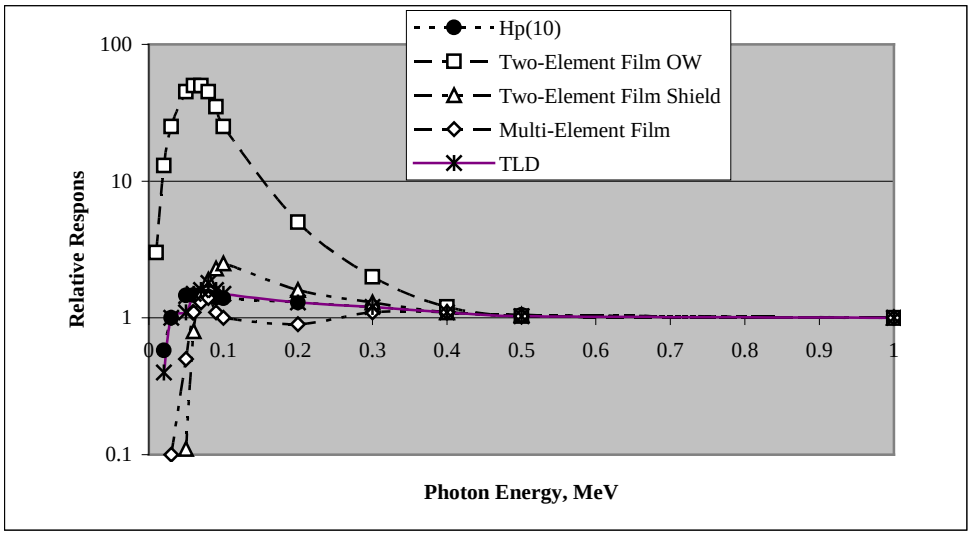

Figure 5.3.2.1-1 Estimated SRS Dosimeter Photon Response Characteristics

To evaluate the dosimeter response to lower-energy (i.e., < 100 keV) photons that are significant in plutonium facilities, Hanford conducted intercomparison testing of all Hanford historical dosimeter film designs (Wilson et al 1990) using A-P irradiations only. Although there are differences in films and particularly the filters used in the multi-element dosimeters, good comparison in energy response for SRS and Hanford dosimeters is expected. The results of this testing is summarized in Table 5.3.2.1-2. The dosimeter results for energies greater than 100 keV, are consistent with the IARC results showing an over-estimate of $H_p(10)$ for the two-element dosimeter. For energies less than 100 keV, the two-element dosimeter underestimates the photon dose. This under-response is also illustrated in the original University of Chicago two-element dosimeter energy response curve (Pardue et al 1944).

Table 5.3.2.1-1 IARC Testing Results for US beta/photon Dosimeters.

		118 keV		208 keV		662 keV	
Geometry	**Phantom**	Mean[(a)]	SD/ Mean	Mean[(a)]	SD/ Mean	Mean[(a)]	SD/ Mean
US-2 (Two-Element Film Dosimeter)							
A-P	Slab	3.0	2.1	1.3	1	1.0	0.8
A-P	Anthropomorphic	3.0	4.2	1.2	1.9	1.0	1.8
Rotational	Anthropomorphic	2.2	2	1.4	3	1.2	3.2
Isotropic	Anthropomorphic	1.5	4.4	1.1	1.6	1.0	2.7
US-8 (Multi-Element Film Dosimeter)							
A-P	Slab	1.0	1.5	1.0	0.8	0.8	1.7
A-P	Anthropomorphic	0.8	9.5	0.9	6	0.8	1.8
Rotational	Anthropomorphic	1.2	1.9	1.2	17	1.1	1.8
Isotropic	Anthropomorphic	1.0	3	1.2	9	1.0	2.3
US-22 (Multi-Element Thermoluminescent Dosimeter)							
A-P	Slab	0.9	4.4	0.9	3.9	0.9	3.5
A-P	Anthropomorphic	0.8	3.1	0.9	2.1	0.9	3.9
Rotational	Anthropomorphic	1.1	3.1	1.2	1.5	1.0	4.1
Isotropic	Anthropomorphic	0.9	0.3	1.0	2.5	0.9	1.6

(a) Ratio of recorded dose to $H_p(10)$.

SRS adopted a conservative practice**1** historically to calculate the deep dose for selected workers in plutonium facilities in specific instances when gonadal dose was considered significant. This practice involved adding the open window (i.e., skin dose), based on a 16 keV k-fluorescent calibration, to the deep dose as measured from the film response behind the 1 mm silver filter. Figure 5.2.3.1-1 illustrates the potential significance of this correction to the routinely recorded dose. Based on this practice, routine SRS deep dose estimates are judged to be reasonably accurate compared to Hp(10) for all photon energies. SRS used uranium to routinely calibrate the dosimeter open window dose response."

Table 5.3.2.1-2 Testing Results for Hanford Two-Element and Multi-Element Film Dosimeters for Energy and Angular Response[a,b]

	Anterior-Posterior (AP) Exposure			Rotational Exposure		
	Film Dosimeters			Film Dosimeters		
Beam (Energy, keV)	**Two Element 1944-56**	**Multi-Element, 1957-71**	**TLD, 1972-Present**	**Two Element, 1944-56**	**Multi-Element, 1957-71**	**TLD, 1972-93**
16[b]	0.1	0.9				
59[b]	0.5	1.1				
M150(70)	0.7	0.70	0.95	1.31	1.31	1.77
H150(120)	1.6	0.64	0.87	3.00	1.20	1.64
^{137}Cs(662)	1.0	1.0	1.0	1.46	1.46	1.46

(a) Divide recorded dose by table value to estimate $H_p(10)$.
(b) Based on Wilson et al (1990).

5.3.2.2 *Neutron Dosimeters*

Two general types of SRS neutron dosimeters have been used and these differ dramatically in their response to neutron radiation.

NTA - The nuclear track emulsion, type A, (NTA) neutron dosimeter consisted of the NTA film enclosed in the same holder as used for SRS beta/photon dosimeters until November 1959 when a NTA specific dosimeter holder was implemented (Taylor et al 1995).

TLD - The SRS TLD neutron (TLND) dosimeter was implemented on 1 January 1971. The TLND is a sphere design, using two polyethylene spheres covered on the curved surface with cadmium. The dosimeter is worn using a belt to minimize distance from the worker's body (Hoy 1972). SRS implemented a commercial Panasonic neutron TLD system on January 1, 1995.

The AEC held a series of Personnel Neutron Dosimetry Workshops to address problems being experienced by the respective AEC sites concerning accurate measurement of neutron dose. The first workshop was held 23–24 September 1969 (Vallario et al 1969) with the stated concern: "...for intermediate energy .....neutron sources, NTA personnel neutron dosimeters cannot be effectively used. This leaves a gap in the personnel dosimetry program which at many installations may be quite serious." The workshops

1 NIOSH Procedures manual…. And discussions during June 19, 2003 teleconference among NIOSH, NIOSH contractor and SRS technical staff.

were generally limited to representatives from sites with active personnel neutron dosimetry programs and continued for many years. The 11th DOE Workshop on Personnel Neutron Dosimetry was held in 1991 (Rabovsky et al 1991). The significance of the underestimated neutron dose became evident with the studies leading to implementation of the thermoluminescent dosimeters (TLDs) and using simultaneous NTA and TLD (neutron) measurements of recorded worker dose. Hanford (Nichols et al 1972) conducted field studies with workers wearing both types of dosimeters.

Response characteristics of the SRS NTA type dosimeter and the TLND neutron dosimeter in comparison with other DOE neutron dosimeters are presented in Brackenbush et al (1980). In general, the response of the NTA dosimeter decreases with decreasing neutron energy that are greater than a threshold energy estimated by SRS to be about 500 keV (Taylor et al 1995) and the TLD response increases with decreasing neutron energy. The response of both dosimeters is highly dependent upon the neutron energy spectra. Both dosimeter types require matching the laboratory calibration neutron spectra with the workplace spectra for accurate results. Results reported by Hoy during the first AEC Neutron Dosimetry Workshop in 1969 indicated that simultaneous laboratory2 measurements of the dose equivalent with NTA was about one-half to one-fourth that compared with other methods including the TLND (Vallario et al 1969).

5.3.3 Calibration

Potential error in recorded dose is dependent on the dosimetry technology response characteristics to each radiation type, energy, and geometry; the methodology used to calibrate the dosimetry system; and the extent of similarity between the radiation fields used for calibration and that present in the workplace. The potential error is much greater for dosimeters with significant variations in response such as film dosimeters for low energy photon radiation and both the NTA and TLND for neutron radiation.

5.3.3.1 *SRS Beta/Photon Dosimeters*

SRS dosimeters were originally calibrated using primarily uranium and ^{226}Ra using in-air (i.e., no phantom) exposures to selected levels. K-fluorescent x-rays were used to develop dosimeter response characteristics for the lower energy photon fields in plutonium facilities. This practice is similar to other AEC sites. Taylor et al (1995) describes adjustments to SRS recorded dose to estimate Hp(10) based on SRS preparations for DOELAP performance testing in the mid-1980s. At that time, it was concluded that:

- Prior to 1 January 1986 the recorded dose of record (i.e., photon) dose should be multiplied by a factor of 1.119 (11.9%).

- Prior 1 January 1987 recorded dose of record (i.e., photon) should be multiplied by a factor of 1.039 (3.9%).

These changes in the recorded dose should be made to arrive at an assured claimant-favorable treatment. Common sources of laboratory bias are shown in Table 5.3.3.1-1

2 Based on June 19, 2003 teleconference between NIOSH, NIOSH project contractors and SRS technical staff.

for personnel beta/photon dosimeter calibration based on comparison of the recorded dose with Hp(10).

5.3.3.2 *SRS Neutron Dosimeters*

Historical aspects of the calibration of SRS NTA and TLNDs are described by Taylor et al (1995) and in articles by Hoy that are included in several reports during Atomic Energy Commission workshops on personal neutron dosimetry such as the first during 23-24 September 1969 (Vallario et al 1969) and second during 8-9 July 1971 (Vallario et al 1971). Table 5.3.3.2-1 describes the common sources of laboratory bias for personnel neutron dosimeter calibration based on expected comparison of the recorded dose with Hp(10).

Table 5.3.3.1-1 Common Sources of Laboratory Bias for Beta/photon Dosimeter Calibration Parameters[a]

Parameter	Historical Description	Anticipated Laboratory Bias[b]
In-Air Calibration	In the 1980s, SRS began exposing calibration dosimeters on phantom (used to simulate worker's body). As such, previous calibrations do not include response from radiation backscatter response.	Recorded dose of record **too high**. Backscatter radiation from worker's body is highly dependent upon dosimeter design.
Radiation Quantity	Prior to the 1980s, SRS dosimeter systems were typically calibrated to photon beam measured as *exposure*.	For higher energy ^{226}Ra and ^{137}Cs gamma radiation used to calibrate dosimeters this caused only a slight (about 3%) **under-response** in recorded dose.
Tissue depth of dose	Historically, SRS used selected depths of 1 or 2 cm to estimate the deep dose.	The numerical effect of this for photon radiation is comparatively low. SRS dosimeter designs had filtration density thickness of about 1,000 mg/cm^2 that would relate to the 1 cm depth in tissue.
Angular Response	SRS dosimeter system is calibrated using anterior-posterior (A-P) laboratory irradiations.	Recorded dose of record likely **too low.** Dosimeter response is lower at non A-P angles. Effect is highly dependent upon radiation type and energy.
Environmental Stability	SRS film dosimeter and TLD systems are subject to signal fade with time, heat, humidity, light, etc.	Recorded dose of record likely **too low** however this depends strongly upon when calibration dosimeters are irradiated. Mid-cycle calibration minimizes effects.

a. Judgment based on SRS dosimeter response characteristics.
b. Recorded dose compared to $H_p(10)$.

Table 5.3.3.2-1 Common Sources of Laboratory Bias for Neutron Dosimeter Calibration Parameters[a]

Parameter	Historical Description	Anticipated Laboratory Bias[b]
Source Energy Spectra-	In the 1980s, SRS began exposing calibration dosimeters on phantom (used to simulate worker's body). As such, previous calibrations do not include response from radiation backscatter response. The beam was also degraded in an attempt to better represent the workplace.	The delivered dose was **uncertain** as noted in Brackenbush et al (1987). (see workplace radiation fields)
Radiation Quantity	Neutron dose quantities used to calibrate SRS neutron dosimeter systems have varied historically. For many years the neutron *dose equivalent index* was used.	The effects of the respective dose quantities used to calibrate SRS neutron dosimeters is **uncertain** and could be evaluated in comparison to the Hp(10) dose used in DOELAP performance testing.
Angular Response	SRS dosimeters calibrated using anterior-posterior (A-P) laboratory irradiations.	Recorded dose of record likely **too low** since dosimeter response is lower at angles other than A-P. Effect is highly dependent upon energy.
Environmental Stability	SRS NTA film dosimeter and TLD systems are subject to signal fade with time, heat, humidity, light, etc.	Recorded dose of record likely **too low** because of fading however this effect depends strongly upon when calibration dosimeters are irradiated.

a. Judgment based on SRS dosimeter response characteristics.
b. Recorded dose compared to $H_p(10)$.

5.3.4 Workplace Radiation Fields

SRS operations are characterized by significant complex beta, photon and neutron radiation fields in SRS reactor, irradiated fuel processing, plutonium handling, neutron source production, and radioactive waste facilities. Descriptions of processes and primary nuclides are presented in the respective SRS External Dosimetry Technical Basis documents (WSRC 2003).

5.3.4.1 *SRS Workplace Beta/Photon Dosimeter Response*

Essentially all SRS radiological work areas involved beta/photon radiation covering a wide range of energies characteristic of the radionuclides being handled within the respective facilities and processes. Radiation beta/photon fields characteristic of SRS facilities can be generally defined based on historical information as presented in Table 5.3.4.1-1. The expected performance of SRS dosimeters to these fields has been estimated in this TBD using the extensive evaluations of the very similar Hanford dosimeters (Wilson et al 1990, Fix et al 1994, 1997), the IARC intercomparison study results (Thierry-chef et al 2002) and available internal3 SRS documentation

3 For example, letter, dated September 3, 1968 from W.R. Tyson, to J.D. Ellett entitled, "Personnel Radiation Exposures – Building 235-P."

Table 5.3.4.1-1 Selection of Beta and Photon Radiation Energies and Percentages for SRS Facilities

Process/ Buildings	Description	Operations		Radiation Type	Energy Selection	Percent
		Begin	End			
Fuel Fabrication	Produced reactor fuel and target assemblies.	1953	1989	Beta	> 15 keV	100%
300M	Uranium fuel fabrication	1953	1989	Photon	30 – 250 keV	100%
Reactors	**During Operation:** Highly dispersed fields of higher-energy photon radiation fields from fission process, activation/ fission product nuclides.. Potential for significant airborne nuclides and there may be significant higher-energy beta radiation.			Beta	> 15 keV	100%
	While not in Operation: Highly dispersed fields of higher-energy photon radiation fields from activation/fission product nuclides.. There may be significant higher-energy beta radiation during maintenance work resulting from fission products.					
105C	Heavy Water Moderated Reactors	3/1955	6/1985	Photon	30 – 250 keV	50%
105K		10/1954	1988		> 250 keV	50%
		1992	7/1992			
105L		7/1954	1968			
		1985	6/1988			
105P		2/1954	8/1988			
105R		12/1953	6/1964			
Processing Plants	**Radiochemical Operations:** Highly dispersed fields of higher-energy photon radiation fields from activation/fission product nuclides dominant to most exposure profiles. Potential for higher-energy beta radiation during sampling and, maintenance work resulting from fission products.			Beta	> 15 keV	100%
200F	Radiochemical processing plant	1954		Photon	30 – 250 keV	50%
200H	Radiochemical processing plant	1955			> 250 keV	50%
Plutonium Production	**Plutonium Component Production**: Plutonium is machined into weapon components using a glove box assembly process with predominant close anterior exposure to workers. Radiation characteristics in this area involve significant lower-energy photons and lower-energy neutron radiation.			Photon	< 30 keV	25%
	Plutonium Storage: Radiation characteristics in this area generally involve dispersed lower-energy neutron radiation and scattered photons to include 60 keV Am-241 gamma ray.				30 – 250 keV	75%
221F, B-line	Plutonium Finishing Process	1954				
221H, B-line	Plutonium Finishing Process	1955				
772F	Production Control Laboratory	1955				
235F	Plutonium Fuel Facility	1955				
736A	Calibration Facility	1953				
Radionuclide Production	Radiation characteristics are highly dependent upon the radionuclide being produced. SRS has produced several radionuclides including tritium, ^{238}Pu, ^{239}Pu, americium, curium, californium, etc.			(Specific to radio-nuclide)		
773A	Savannah River Laboratory	1953		Photon, default	30 – 250 keV > 250 keV	50% 50%
Waste Handling	Radiation characteristics are highly dependent upon the source of waste but typically fission product nuclides (^{90}Sr/^{90}Y, ^{137}Cs) are dominant.			Beta	> 15 keV	100%
200F	General waste handling, storage tanks, etc.	1953	Ongoing	Photon	30 – 250 keV	50%
200H	General waste handling, storage tanks, etc	1953	Ongoing		> 250 keV	50%

Common beta/photon personnel dosimeter parameters important to Hp(10) performance in the workplace are summarized in Table 5.3.4.1-2. Based on the collective information, SRS dosimeters are expected to reasonably measure the Hp(10) dose under all SRS workplace radiation fields. The SRS historical practice, for the two-element film dosimeter in plutonium facilities characterized by predominant photon energies <100 keV, to calculate the total whole body deep dose by summing the shallow dose from the open window film response based on a 16 keV fluorescent x-ray calibration and the deep dose from the 1 mm silver filtered film response based on a ^{226}Ra calibration, will result in a over-estimate of the actual Hp(10) dose. In addition, laboratory irradiations of the two-element dosimeter have shown an over-response of the actual Hp(10) dose by about a factor of 2 to photons greater than 100 keV. A claimant-favorable approach is proposed to ignore this over-response because of the complexity of workplace photon energies and exposure geometries that tend to result in an under-estimate of the Hp(10) dose. The respective SRS dosimeters have filtration of approximately 1000 mg/cm^2 (i.e., nearly equivalent to 1 cm depth in tissue) for those regions of the dosimeter used to measure the whole body dose. The response to beta radiation in SRS workplaces is limited because beta radiation cannot penetrate the filtration. As such, a reasonable estimate of deep dose, compared to Hp(10), is expected for SRS beta/photon workplace radiation.

Table 5.3.4.1-2 Common Workplace Beta/Photon Dosimeter $H_p(10)$ Performance[(a)]

Parameter	Description	Workplace Bias[(b)]
Exposure Geometry	SRS dosimeter system calibrated using anterior-posterior (A-P) laboratory irradiations.	Recorded dose of record likely **too low** since dosimeter response is lower at angles other than A-P. Effect is highly dependent upon radiation type and energy.
Energy Response	SRS film "deep dose" dosimeter response to photon radiation <100keV is too low and >100keV is too high.	Bias in recorded dose **depends upon the photon spectra in the workplace**. SRS practice to include the shallow dose based on a 16-keV calibration to the deep dose for selected workers in SRS plutonium facilities where gonadal dose was significant eliminated this common source of potential under-response. Reasonable estimate of Hp(10) dose is expected.
Mixed Fields	SRS dosimeters respond to beta and photon radiation.	Filtration of about 1,000 mg/cm^2 over dosimeter component used to calculate deep dose will minimize dosimeter response to beta radiation. Reasonable estimate of Hp(10) dose is expected.
Missed Dose	Doses less than Minimum Detection Level recorded as zero dose.	Recorded dose of record likely **too low** and the relative impact is significant primarily in earlier years with frequent dosimeter exchange and film dosimeters with higher MDLs.
Environmental Effects	Workplace heat, humidity, etc., fades dosimeter signal.	Recorded dose of record likely **too low.**

a. Judgment based on SRS dosimeter response characteristics and workplace radiation fields.
b. Recorded dose compared to $H_p(10)$.

5.3.4.2 *SRS Workplace Neutron Dosimeter Response*

There are four main areas at SRS where there is a potential for neutron exposure: the plutonium facilities in the 200 area; the Calibration Facility (736-A) and the Cf-252 facility (773-A) in 700 area; the reactor areas (100) during operation; and the 321 Building (Pu-Al alloys) in the 300 Area. Each of these facilities has different neutron exposure spectrum. These areas also have demonstrated different neutron to photon ratios. The following sections discuss the different energy spectra and neutron to photon ratios by facility.

5.3.4.2.1 *Plutonium Facilities*

The majority of the neutron exposure at the Savannah River Site (SRS) is associated with the separation, finishing, and storage of plutonium. These operations are performed primarily on the 200F B-line (221-F), the 200H HB-line (221-H), the Production Control Laboratory (772-F), and at the PuFF Facility (235-F). Neutron dose is also associated with process control analysis and storage of the product. The plutonium is introduced into the SRS plutonium finishing processes as plutonium fluoride. The plutonium fluoride is converted into plutonium oxide and made into buttons. Plutonium fluoride produces an (α,n) reaction resulting in a significant source of neutrons. Neutrons are also emitted spontaneously from ^{238}Pu and ^{240}Pu, which compose a large activity fraction of the plutonium buttons. Plutonium-238 is the primary isotope used for the production of heat sources (Bebbington, 1990).

5.3.4.2.1.1 Neutron Energy Spectra

Brackenbush et al (1987) measured neutron spectra at several locations across the Savannah River Site. Figure 5.3.4.2-1 provides the neutron spectra measured at glove boxes on the FB and HB lines respectively. It should be noted that the data in Brackenbush et al. (1987) were depicted as dose equivalent rates, however for simplicity of calculation, a one hour exposure was assumed in order to use dose equivalent.

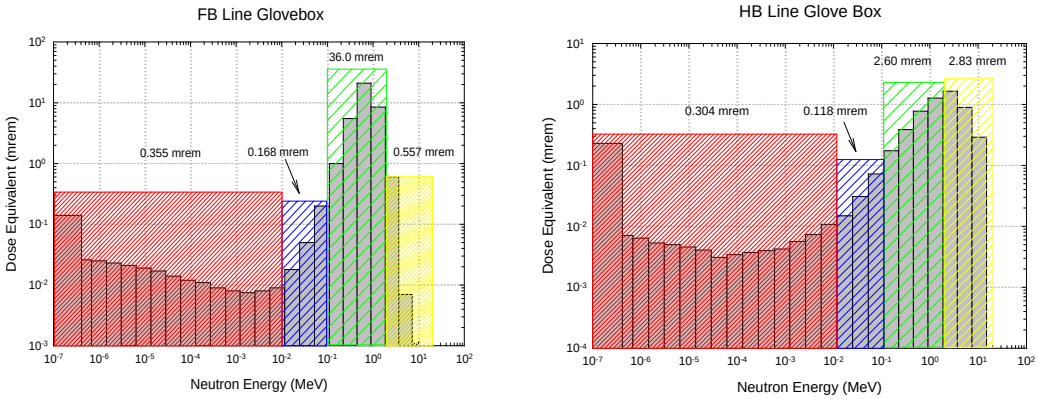

Figure 5.3.4.2-1 Neutron Energy Spectra Recorded at SRS Plutonium Facilities with Neutron Energy Groups Overlay.

Although somewhat different neutron spectra were recorded for the different exposure areas (HB Line, FB Line, and storage facility), similarities were observed in the general

shape of the spectra and the energy of the flux peak at approximately 1-2 MeV. The HB glovebox spectra in Figure 5.3.4.2-1 demonstrated a higher peak energy which is possibly the result of more spontaneous fission from Pu-238 but other spectra from the HB line indicated a spectra with neutron energies between the FB Glove box line spectra and the HB Glovebox spectra depicted.

The fraction of the total dose in each neutron energy group was determined by subdividing the neutron spectra into the four lower neutron energy groups discussed in the External Dose Reconstruction Implementation Guideline (NIOSH 2002) (See overlays in Figure 5.3.4.2-1. The highest neutron energy group (>20 MeV) was not used since operations at SRS did not produce neutrons of this energy. The dose for each neutron energy group was calculated by multiplying the neutron flux (Ø) provided in Brackenbush et al (1987) by the corresponding flux to dose-rate conversion factors (DCF) found in NCRP 38. The neutron doses in each NCRP 38 energy interval are then summed to develop the four neutron group doses. The dose fraction (D_f) for each neutron energy group (n) was then calculated by dividing the neutron group dose by the total dose (D_T) (Eqn 5.1).

$$D_f(E_n) = \frac{\sum_i \phi(E_i) DCF_i}{D_T} \qquad (5.1)$$

where:
$Ø(E_i)$ = Neutron flux of the ith energy bin
DCF_i = NCRP 38 (1971) flux to dose-rate conversion factor for the ith energy bin
D_T = Total dose

Tables 5.3.4.2.1.1-1 and 5.3.4.2.1.1-2 provide the neutron dose fractions by energy group using the data measured by Brackenbush et al (1987).

Table 5.3.4.2.1.1-1 Measured Dose Fractions at Different Locations in H Area

Neutron Energy Group	HB-Line Glovebox	HB-Line in Front of Hood	Through Glove Port (extremity)	Default
< 10 keV	0.041	0.003	0.028	
10 – 100 keV	0.026	0.004	0.002	
0.1 – 2 MeV	0.438	0.584	0.563	
2 – 20 MeV	0.495	0.409	0.407	
Claimant Favorable Dose Fractions				
0.1 – 2 MeV	0.5	0.6	0.6	0.6
2 – 20 MeV	0.5	0.4	0.4	0.4

Table 5.3.4.2.1.1-2 Measured Dose Fractions at Different Locations in F Area

Neutron Energy Group	FB-Line Glovebox	Storage Area	Default
< 10 keV	0.009	0.029	
10 – 100 keV	0.007	0.017	
0.1 – 2 MeV	0.966	0.816	
2 – 20 MeV	0.018	0.138	
Claimant Favorable Dose Fractions			
0.1 – 2 MeV	1.0	0.86	1.0
2 – 20 MeV		0.14	

Although Brackenbush et al. (1987) measured some dose from lower (<10 keV) and intermediate (10-100 keV) energy neutrons, the contribution to the total dose was less than 5%, respectively. The Radiation Effectiveness Factors used in the Interactive RadioEpidemiological Program (IREP) to calculate the Probability of Causation (PC) for these two neutron energy groups are less than the fission spectrum energy group (0.1 – 2 MeV) (Kocher et al., 2002). As a result combining the lower and intermediate energy neutron groups into the fission spectrum group is a reasonable and claimant favorable simplification of the dose calculation. In both table 5.3.4.2.1.1-1 and 5.3.4.2.1.1-2, claimant favorable default values are also provided when there is insufficient work history information to determine the type of operation the energy employee conducted.

5.3.4.2.1.2 Neutron to Photon Ratio

Trends in the annual SRS and Hanford neutron collective dose (Taylor et al 1995, Buschbom and Gilbert 1993, respectively), normalized to the annual plutonium production (DOE 1996), are illustrated in Figure 5.3.4.2-2. It is evident in this figure that the collective neutron dose was under-recorded prior to implementation on 1 January 1971 of the SRS TLND and 1 January 1972 for the Hanford TLD. The extent of the under-estimate is difficult to estimate. SRS and Hanford showed a significant increase in the ratio of the annual collective neutron dose to the annual plutonium production when the TLD neutron dosimeters were implemented. The continued increase in the ratio shown in Figure 5.3.4.2-1 may be attributable to the increasingly lower handling of plutonium as done during production compared to increasing work activities in later years to store plutonium with the associated high scatter of neutrons from shielding. The TLND is well known to over-respond to lower energy neutrons (Scherpelz et al 2002).

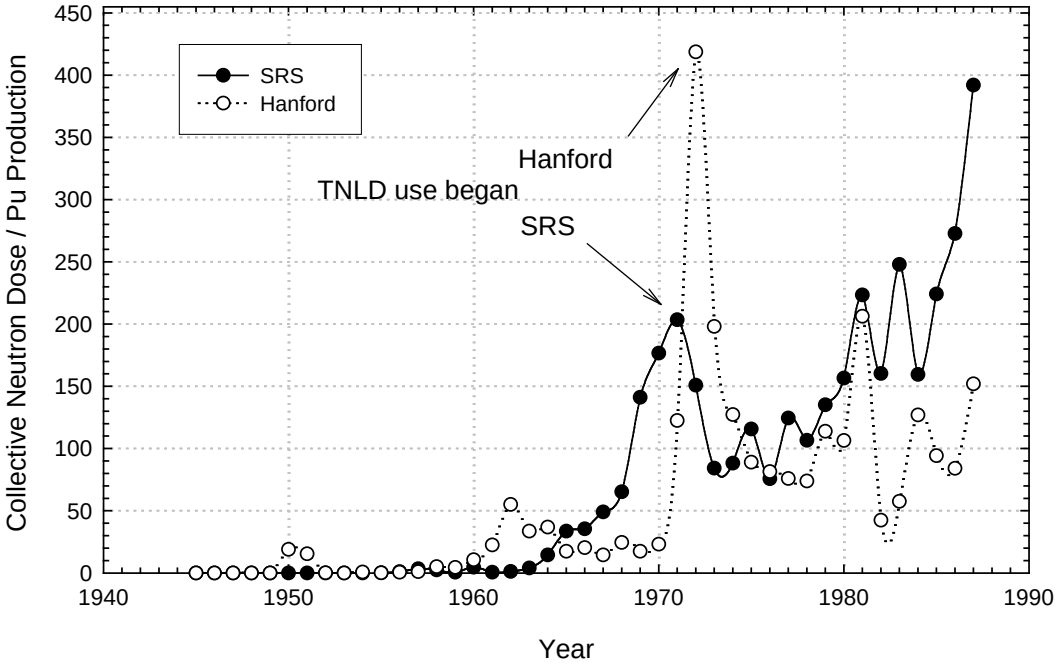

Figure 5.3.4.2-2 Trends in SRS and Hanford Collective Neutron Dose Normalized to Plutonium Production.

Subsequent to the implementation of the Hanford TLD on 1 January 1972, AEC headquarters staff conducted a detailed review of recorded neutron dose for Hanford personnel using a committee of technical experts from Hanford, SRS, and other AEC facilities (Biles 1972). Central to this investigation was the selection of 18 long-term Hanford workers for detailed evaluation. Fix et al (1996) conducted a later analysis of this information using additional dosimetry data recorded after 1972. Three distinct periods of dose recording from 1950 through the present are identified in Table 5.3.4.2.1.2-1 corresponding to the respective Hanford two-element, multi-element and thermoluminescent dosimeters that parallel similar dosimeters implemented at SRS.

Table 5.3.4.2.1.2-1 Recording Periods for Selected Hanford Plutonium Workers

Category	Period	Description
A	1950-56	This period of record involved use of the original Hanford two-element dosimeter for nonpenetrating (shallow) and penetrating (deep) dose components and NTA film for neutron radiation.
B	1957-71	This period of record involved use of the Hanford multi-element film dosimeter for nonpenetrating, x-ray and penetrating dose components, and NTA film for neutron radiation.
C	1972-95	This period of record involved use of the Hanford TLD for beta, photon, and neutron dose components.

All of these workers had the Hanford plutonium facility as their primary work area at least during the 1970s. Nine of the 18 workers have dose histories that extend from 1950 or earlier through 1980 or later. It is interesting to examine trends in this data. For example, the ratios of the shallow to deep dose and neutron to deep dose for the respective recording periods are summarized in Table 5.3.4.2.1.2-2. These ratios exhibit

the expected response characteristics of the respective dosimeters based on laboratory studies.

Table 5.3.4.2.1.2-2 Hanford Ratio of Recorded Dose Components

Recording Period	Ratio (Range)	
	Shallow/Deep	Neutron/Deep
1950-56	1.6 (1.1 - 3.7)	0.003 (0 - 0.06)
1957-71	1.2 (1.1 - 1.7)	0.4 (0.1 – 0.7)
1972-95	1.3 (1.1 – 1.5)	0.6 (0.1 – 1.6)

The shallow dose response for the two-element film dosimeter used at Hanford prior to 1957 is expected to be too high because of the significant over-response of this dosimeter to the low-energy photons prevalent in plutonium facilities (Figure 5.3.4.2-2). These data also show increasing levels of recorded neutron dose, relative to the deep dose, for each succeeding dosimeter design.

While no formal report has been published establishing a neutron to photon ratio for Savannah River Workers, a review has been conducted as part of this Technical Basis Document to validate the assumption that the Hanford plutonium worker's neutron to photon ratio was similar to that experienced by Savannah River workers. Neutron to photon ratios were evaluated for nine plutonium workers (5 from F-Area and 4 from H-Area) totaling 38 monitoring years of data post 1971 (Table 5.3.4.2.1.1-3). While the average neutron to photon ratio was similar to the ratio observed by Fix et al (1996), there was a significant difference in the upper bound of the HB-Line. This variation could be the result of the higher specific activity of Pu-238 in the HB line compared to Pu-239 in the FB-line.

Table 5.3.4.2.1.1-3 Post 1971 SRS Ratio of FB and HB Line Recorded Dose Components

Area	Neutron/Deep Ratio (GM, GSD, 95th%)	Ratio (Average, Range)	
		SRS Neutron/Deep	Hanford Neutron/Deep
FB Line	0.36, 2.52, 1.65	0.52 (0.09 – 1.23)	0.6 (0.1 – 1.6)
HB Line	0.91. 2.84, 5.05	1.29 (0.05 – 3.10)	

It is important to note that while an average was reported, the distribution of ratios more closely resembled a log normal distribution than a normal distribution. A distribution fit was conducted using the Kolmogrov –Smirnoff test statistic. The log normal distribution was demonstrated the best fit of the four distributions chosen (normal log normal, triangular, and uniform). For this reason, the geometric mean, geometric standard deviation and the corresponding 95th% is provided in Table 5.3.4.2.1.1-3. When incorporating uncertainty into the dose reconstruction, these values should be used in accordance with guidance on combining uncertainty provided in the External Dose Reconstruction Implementation Guideline (NIOSH 2002).

Additional neutron to photon doses was found in the Monthly Works Technical Report (DuPont, 1965). A survey ratio of 0.66:1 was found for the HB-Line, and a ratio of 1:1 was observed for the 235-F facility during this time frame. Brackenbush et al (1987) reported a neutron to gamma ratio of approximately 2:1. This variation in neutron to photon ratio is encompassed by the relatively large uncertainty distribution observed with worker data.

5.3.4.2.2 Neutron Source Production Facilities (773-A)

A second area of neutron exposures at the Savannah River Site (SRS) resulted from the production of neutron sources such as Cf-252. Following irradiation in the reactors, Cf-252 was separated. Unlike the plutonium facilities in which alpha particle interactions with light materials such as fluorine produce the neutrons (α,n reactions), Cf-252 produces neutrons by spontaneous fission. As a result, the neutron energy spectra is considerably higher than the plutonium facilities.

5.3.4.2.2.1 Neutron Energy Spectra

Measurements made by Brackenbush et al (1987) at the 773-A facility clearly demonstrated the shift in the neutron energy spectra. Using the same methodology described in the plutonium facility section, the neutron dose fractions were developed and are provided in Table 5.3.4.2.2.1-1.

Table 5.3.4.2.2.1-1 Dose Fractions for Various Operations in 773-A Area

Neutron Energy Group	Glovebox	Shipping Cask	Default
< 10 keV	0.018	0.015	
10 – 100 keV	0.010	0.023	
0.1 – 2 MeV	0.419	0.349	
2 – 20 MeV	0.553	0.613	
Claimant Favorable Dose Fractions			
0.1 – 2 MeV	0.45	0.39	0.5
2 – 20 MeV	0.55	0.61	0.5

As with the plutonium facility dose fractions, a claimant favorable default value is provided when there is insufficient information to determine which dose fractions should be used.

5.3.4.2.2.2 Neutron to Photon Ratio

As indicated with the plutonium workers, a review was conducted as part of this Technical Basis Document to evaluate the assumption that the Hanford plutonium worker's neutron to photon ratio was similar to that experienced by Savannah River neutron source production workers. Neutron to photon ratios were evaluated for 6 neutron source production workers from the 773-A area totaling 30 monitoring years of data post 1971 (Table 5.3.4.2.2.2-1). While the average neutron to photon ratio was slightly higher than the ratio observed by Fix et al (1996) for Hanford plutonium workers, there was a factor of 2 increase in the upper range.

Table 5.3.4.2.2.2-1 Post 1971 SRS Ratio of 773-A Recorded Dose Components

Area	Neutron/Deep Ratio (GM, GSD, 95th%)	Ratio (Average, Range)	
		SRS Neutron/Deep	Hanford Neutron/Deep
773-A	0.62, 2.29, 2.41	0.85 (0.10 – 3.83)	0.6 (0.1 – 1.6)

5.3.4.2.3 Reactor Facilities (100 Area)

The neutron dose to workers in SRS reactor facilities is very low in comparison to the measured photon and beta dose received from contaminated equipment and plant areas, spent reactor fuel, activated reactor components, and other areas containing fission and activation produces encountered during plant maintenance. However, there are a few areas around the reactors where a worker can receive neutron exposure, albeit a small fraction of the photon dose, during reactor operations.

5.3.4.2.3.1 Neutron Energy

Brackenbush et al. (1987) recorded measurement data at the K-reactor door during operations is shown in Figure 5.3.4.2-3. The data indicated the lowest average neutron energy (170 keV) of all facilities measured at SRS.

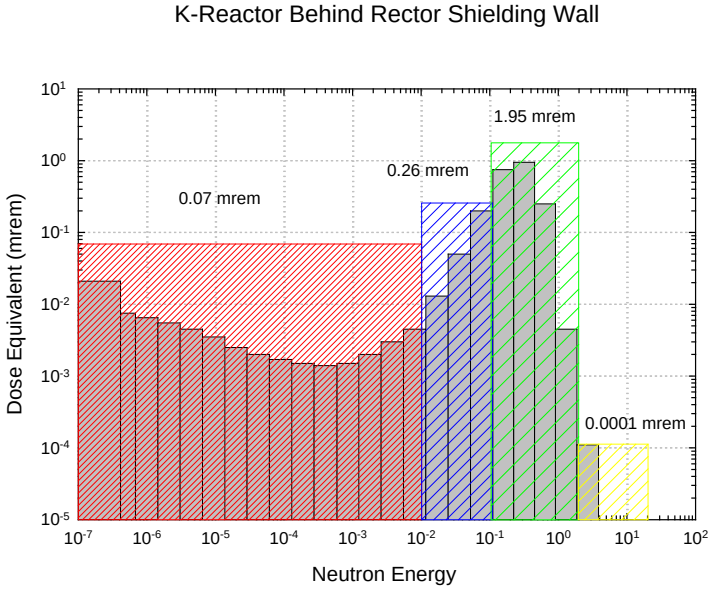

Figure 5.3.4.2-3 Neutron Energy Spectra Recorded Behind the K-Reactor Shield Door with Neutron Energy Groups Overlay.

Table 5.3.4.2.3.1-1 provides the dose fraction per neutron energy group. Unlike the previous two neutron exposure areas, the dose fraction from the intermediate energy neutrons was approximately 12% while the lower energy neutrons remained less than 5%. Since the Radiation Effectiveness Factor (REF) for intermediate energy neutrons is greater than for lower energy neutrons, a claimant favorable simplification is to combine the dose fractions for lower and intermediate energy neutrons.

Table 5.3.4.2.3.1-1 Dose Fractions for Heavy Water Reactors at SRS

Neutron Energy Group	Behind Shield Door	Default
< 10 keV	0.030	
10 – 100 keV	0.115	
0.1 – 2 MeV	0.855	
2 – 20 MeV	0.0001	
Claimant Favorable Dose Fractions		
10 – 100 keV	0.15	0.15
0.1 – 2 MeV	0.85	0.85

5.3.4.2.3.2 Neutron to Photon Ratio

Neutron exposures are expected to be very low compared to beta and photon exposures around the reactors because there is no neutron radiation when the reactors are shutdown for maintenance or refueling when most beta and photon dose to workers is received. Brackenbush et al (1987) reported both the neutron and photon dose rate at the F-Area and the K-Reactor. The neutron to photon dose rate ratio for the F-area was 2:1, while the neutron to photon dose rate ratio for the K reactor was 1:1 or about half of the observed ratio from the F-Area. Using the simple dose-rate ratio reduction factor of 2, the neutron to photon ratio for reactor area workers can be developed based on the personal dose ratio observed for the F-Area workers. Table 5.3.4.2.3.2-1 provides the F-Area neutron to photon ratio and calculated reactor area neutron to photon ratio.

Table 5.3.4.2.3.2-1 Post 1971 SRS Ratio of Plutonium and Reactor Recorded Dose Components

Area	Neutron/photon Ratio (GM, GSD, 95th%)	Ratio (Average, Range)	
		SRS Neutron/photon	Hanford Neutron/photon
Pu Facility	0.36, 2.52, 1.65	0.52 (0.09 – 1.23)	0.6 (0.1 – 1.6)
Reactor	0.18, 2.52, 0.82	0.26 (0.05 – 0.62)	0.26 (0.13 – 0.73)

Fix et al (1996) recalculated neutron doses and evaluated the neutron to photon ratio between the recalculated neutron dose and the photon dose using several methods. Based on this data, the most claimant favorable neutron to photon ratio for 100 area workers at Hanford between 1950 and 1961 ranged from 0.13 to 0.73 with a weighted average of 0.26. The SRS neutron to photon ratio is expected to be considerably lower than that observed from the Hanford reactors because there was minimal neutron radiation in routine work areas during reactor operation due to design (below grade) and shielding. Unlike the graphite reactors, in which workers could conduct work on the face of the reactor during operation, access to the lower plenum where the reactor core and cooling system were located was restricted. In addition, since most of the photon dose was received during refueling operations, increasing the denominator of the ratio, thus further lowering the ratio compared to plutonium facility workers. As a result, the neutron to photon ratio for SRS reactors is considered to be an upper bound of the actual neutron to photon ratio.

Before applying the ratio listed in Table 5.3.4.2.3.1-1, careful consideration should be given to the energy employee's work history. There are occupations around the reactor

areas in which no neutron dose would be expected (office or administrative staff). Generally only workers conducting work near the entrances to the lower plenum (Shield Door), the crane wash area, or the disassembly area have the potential for neutron exposure. There was also some neutron exposure in the assembly area due specific campaigns in which Pu-Al alloy targets were assembled. In the absence of claim – specific data to arrive at a neutron to photon ratio from Table 5.3.4.2.3.2-1 , a neutron to photon ratio of 0.10^4 should be applied to all non-administrative job categories.

5.3.4.2.4 Other Facilities

5.3.4.2.4.1 Calibration Facilities (736-A Area)

The neutron exposure spectra for workers at the SRS calibration Facility cover the entire energy spectra. Although Brackenbush et al (1987) measured multiple spectra at this facility, only two that represent the extremes are discussed in this analysis. The first is the routine calibration of dosimeters using a PuF_4 source in free air. The second is the spectra recorded using a PuBe source at a tank containing 150 mm of D2O. Table 5.3.4.2.3.1-1 demonstrates the wide variation in the resultant dose fractions. Since the relative use of these different sources is not known and the Radiation Effectiveness Factor (REF) is the largest (claimant favorable) for fission spectrum neutrons, the default was chosen based on the PuF_4 source.

The neutron to photon ratio for the calibration facility is not currently known. However, since intermittent exposure is considered the most probable scenario, the neutron to photon ratio is not expected to be greater than that observed at the 773-A facility. A reasonable and claimant favorable assumption would be to use the neutron to photon ratio for the 773-A facility.

Table 5.3.4.2.3.1-1 Dose Fractions at the Calibration Facility (736-A)

Neutron Energy Group	PuF_4 Source w/o Drum	PuBe Source at 150 mm D_2O	Default
< 10 keV	0.008	0.127	
10 – 100 keV	0.007	0.046	
0.1 – 2 MeV	0.819	0.374	
2 – 20 MeV	0.166	0.453	
Claimant Favorable Dose Fractions			
0.1 – 2 MeV			0.83
2 – 20 MeV			0.17

5.3.4.2.4.2 Fuel Fabrication Area (321-M)

For certain campaigns, Pu-Al alloy targets were fabricated in the 321-M area. Figure 5.3.4.2-4 reproduced from Brackenbush et al (1987), compares the neutron spectra of Pu-Al to various other neutron sources. As can be seen from the graph, Pu-Al alloy has a similar neutron energy spectra to PuF4. Since the Pu-Al alloy is similar to a bare PuF4 source, the

4 Claimant favorable factor determined during June 19, 2003 teleconference among NIOSH, NIOSH contractor and SRS technical staff for default use based on minimal neutron dose in SRS reactors, absolutely, and certainly in comparison with the associated photon dose.

dose fraction for the PuF4 source w/o Drum should be used (Table 5.3.4.2.3.1-1). Neutron exposures in the Fuel Fabrication Area are expected to be intermittent and were not continuous over the operation of the SRS, thus a neutron to photon ratio should not be used. Based on dosimetry records from SRS, it appears that during the Pu-Al campaigns, workers were monitored for neutron exposure. Since the Pu-Al alloy spectra is of similar energy to PuF4, approximately 90% of the spectra is greater than the 500 keV threshold for NTA film. As a result, the NTA film measurements are expected to be reasonably accurate to within parameters discussed in the External Dose Reconstruction Implementation Guideline (NIOSH 2002).

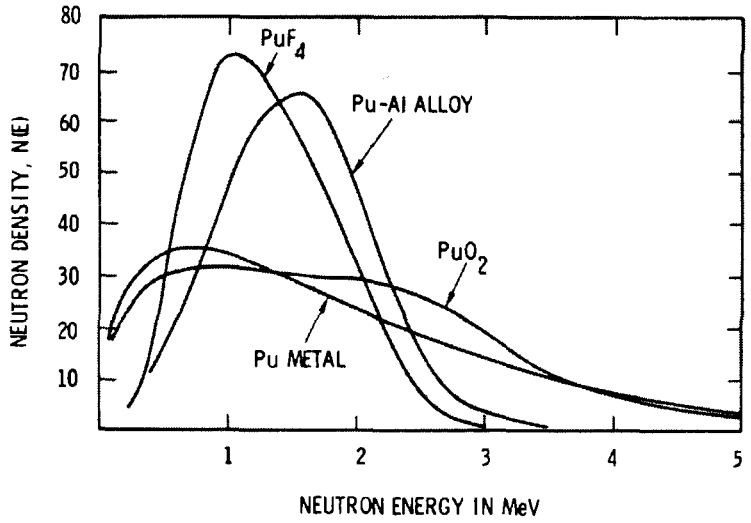

Figure 5.3.4.2-4 Comparison of Various Neutron Spectra from different Plutonium Compounds and Alloys (Brackenbush et al, 1987)

5.3.5 SRS Workplace Neutron Dose Uncertainty

As noted previously, in 1987, SRS contracted the Pacific Northwest National Laboratory (PNNL) to conduct neutron spectra and dose measurements at selected SRS facilities (Brackenbush et al 1987) similar to measurements made by PNNL at Hanford (Brackenbush et al 1991, Endres et al 1996, Scherpelz et al 2000, Fix et al 1981, 1982). These neutron measurements were made using several methods to measure neutron dose including Tissue Equivalent Proportional Counters (TEPCs), portable neutron survey instruments, and the SRS Hoy TLND. In addition, track etch dosimeters from Hanford and Lawrence Livermore National Laboratory (LLNL) were used since they were being studied for potential use in combination personnel neutron dosimeter systems. Energy spectrum measurements were made using multisphere (Bonner) sphere spectrometers, ^{3}HE spectrometers, and NE-213 liquid scintillation spectrometers.

As reported in the PNNL report, measurements with the TEPC, multisphere system and ^{3}HE spectrometer were in general agreement. The TLND agreed within about 30% for most measurement locations along the plutonium production lines and storage areas. The TLND was within a factor of 3 (i.e., 0.3 to 3) for the extremes in neutron energy

spectra encountered at the K-reactor door (i.e., highly thermalized field) and for a californium shipping cast (i.e., where most lower energy neutrons had been removed). Over long time periods, workers would generally be expected to be involved in several different exposure profiles that will serve to minimize the extremes identified. These results are indicative of the technical difficulties to accurately measure neutron dose in the workplace. Table 5.3.4.2-3 presents a summary of common workplace neutron dosimeter performance characteristics.

Table 5.3.4.2-3 Common Workplace Neutron Dosimeter Hp(10) Performance(a)

Parameter	Description	Workplace Bias[(a)]
Workplace neutron energy spectra	SRS NTA dosimeter response decreases and SRS TLD response increases with decreasing neutron energy.	Depends upon workplace neutron spectra. NTA recorded dose of record likely **too low.** Article by Hoy presented in Vallario et al (1969) identified ratios of 2 to 4 under-response.
Exposure Geometry	NTA Dosimeter response increases with increasing exposure angle and TLD response decreases with increasing exposure angle.	NTA recorded dose likely **too high** since dosimeter response is higher angles other than A-P. TLD recorded dose of record likely **too low** since dosimeter response is lower at angles other than A-P. Effect is highly dependent upon neutron energy.
Dosimeter Placement	SRS apparently instructed workers to wear NTA dosimeter facing the workers body to enhance dosimeter response (i.e., to respond to backscatter neutron radiation).	Recorded dose of record likely **too low.** Reflected neutrons are expected to be less-than the NTA energy threshold identified as 0.5 MeV by SRS.
Mixed low energy photon and neutron radiation	SRS NTA dosimeter responds significantly to lower-energy photons that can result in fogging of the NTA film.	NTA recorded dose of record likely too low. SRS NTA dosimeters selected to minimize photon response. SRS TLDs have capability to remove photon dose contribution.

a. Judgment based on SRS dosimeter response characteristics and workplace radiation fields.
b. Recorded dose compared to $H_p(10)$.

Taylor et al (1995) reported that SRS participated in Personal Dosimetry Intercomparison Studies starting in 1974. SRS dosimetry demonstrated an average error of approximately 40% in the first intercomparison. According to Taylor et al (1995), a marked improvement was noted and the average error was 13.9% in 1985. The SRS site has noted a trend of improving neutron dosimetry through the years. Measurements of TLND performance at SRS in 1987 (Brackenbush et al., 1987) indicate that the SRS measured neutron dose with the TLND (beginning 1 January 1971) is reasonably correct. For dose reconstruction under EEOICPA, a claimant favorable standard error estimate of 50% should be made for neutron dosimetry between 1971 and 1985. Starting in 1985, a standard error of 15% should be assumed in the neutron recorded dose.

5.4 Adjustments to Recorded Dose

Adjustments to the SRS reported dose are necessary to arrive at a claimant-favorable dose considering the uncertainty associated primarily with the complex SRS workplace radiation fields and exposure orientations.

5.4.1 Photon Dose Adjustments

Taylor et al (1995) identified calibration corrections to the reported SRS TLD-Deep photon dose for the periods prior to 1986 and 1987, respectively, to estimate the Hp(10) dose. Photon dose corrections summarized in Table 5.4.1-1 are necessary to calculate an adjusted dose that is claimant favorable considering historical technological limitations of SRS dosimeters. For the years 1987 to the present, no correction is needed, the reported dose is Hp(10) equivalent.

Table 5.4.1-1 Adjustments to Reported SRS Deep Photon Dose

Time Period	Dosimeter	Facility	Step	Adjustment to Reported Dose
Prior to 1/1/1986	All beta/photon dosimeters	All facilities	A	Multiply reported TLD-Deep photon dose by a factor of 1.119 to estimate Hp(10).
For the year 1986	TLD beta/photon dosimeter	All facilities	B	Multiply reported TLD-Deep photon dose by a factor of 1.039 to estimate Hp(10).

5.4.2 Neutron Dose Adjustments

The Savannah River Site (SRS) incorporated the energy variation of the dose equivalent into their calibration methodology. As a result, the recorded dose equivalent (DE_R) is a combination of all neutron energies. In order to calculate the probability of causation, the recorded neutron dose must be separated into neutron energy groups as discussed in section 5.3.4.2 and subsequently converted to ICRP 60 (1990) methodology.

5.4.2.1 *Neutron Weighting Factor*

Adjustment to the neutron dose is necessary to account for the change in neutron quality factors between historical and current scientific guidance as described in NIOSH (2002). At SRS, a single quality factor was not used, but rather a continuum of quality factors based on neutron energy that varies up to 11 at 1 MeV (Singh 2002). The quality factor was incorporated into the calibration methodology which used flux to dose-rate conversion factors for varying neutron energies (Brackenbush 1987, Taylor 1995). According to site personnel (Crase 2003), the SRS flux to dose-rate conversion factors were based on NCRP 38 (1971). NCRP 38 (1971) lists both flux to dose-rate conversion factors and associated quality factors that vary from 2 at energies less than 1 keV to 11 at 1 MeV. In order to convert from NCRP 38 quality factors to ICRP 60 (1990) radiation weighting factors, a curve was fit describing the neutron quality factors as a function of neutron energy. The average quality factor for each neutron energy group was developed by integrating the area under curve and dividing by the neutron energy range as shown in equation 5.2.

$$\overline{Q}(E_{n,0.1-2.0MeV}) = \frac{\int_{0.1}^{2.0} Q_f(E)dE}{Range(2.0-0.1)} \qquad (5.2)$$

Table 5.4.2-1 provides a summary of historical changes in the quality factors and the average NCRP 38 quality factor for the neutron energy groups used in dose reconstruction.

Table 5.4.2-1 Historical Neutron Quality or Weighting Factors

Neutron Energy (MeV)	Historical Dosimetry Guideline[a]	NCRP 38 Quality Factors[b]	Average Quality Factor used at SRS	ICRP 60 Neutron Weighting factor, w_r [c]
2.5×10^{-8}	3	2	2.35	5
1×10^{-7}	10	2		
1×10^{-6}		2		
1×10^{-5}		2		
1×10^{-4}		2		
1×10^{-3}		2		
1×10^{-2}		2.5	5.38	10
1×10^{-1}		7.5	10.49	20
5×10^{-1}		11		
1		11		
2		10	7.56	10
2.5		9		
5		8		
7		7		
10		6.5		
14		7.5		
20		8	Not applicable	5
40		7		
60		5.5		

a. Trilateral meeting in 1949 radiation protection guidelines (Fix et al 1994).
b. Recommendations of NCRP 38.
c. ICRP 60 (1990)

5.4.2.2 *Neutron Correction Factor*

Table 5.4.2-1 provides the average quality factor for the four neutron energy groups which encompass SRS neutron exposures. The neutron dose equivalent correction factor can be calculated by dividing the dose fractions from section 5.3.4.2 for each neutron energy group ($D_f(E_n)$) by the corresponding energy specific average NCRP 38 quality factor ($Q(E_n)$) and then multiplying by the ICRP 60 radiation weighting factor (w_R) as shown in equation 5.3.

$$C_f(E_n) = \frac{D_f(E_n)}{\overline{Q}(E_n)} \times w_R \qquad (5.3)$$

Table 5.4.2.2-1 summarizes the default neutron dose distributions by energy for each neutron exposure area. By multiplying the recorded neutron dose by the area specific correction factors the neutron dose equivalent is calculated. For example consider a 1000 mrem recorded neutron dose by an energy employee working at the HB-Line, the corrected neutron dose is 1144 mrem from neutrons between 0.1-2.0 MeV ***and*** 529 mrem from neutrons with energy between 2 and 20 MeV. Thus the corrected neutron dose is a total of 1673 mrem. These corrections should be applied to measured dose, missed dose and dose determined based on a neutron to photon ratio. Table 5.4.2.2-1 summarizes the default dose fractions by energy and the associated ICRP 60 correction factors for each neutron exposure area.

Table 5.4.2.2-1 Summary of Dose Fractions and associated ICRP 60 correction factors

Process	Description / Buildings	Operations Begin	Operations End	Neutron Energy	Default Dose Fraction (%)	ICRP 60 Correction Factor
Fuel Fabrication	Produced reactor fuel and target assemblies. Intermittent production mixed oxide fuel and plutonium – aluminum (Pu-Al) alloy targets.					
	(321-M)			0.1-2 MeV 2-20 MeV	83% 17%	1.58 0.23
Reactors	During Reactor Operation: Low level neutron exposure through shielding. In addition, some intermittent neutron exposure from mixed oxide fuel and targets.					
	(105C)	3/1955	1/1987	10-100 keV 0.1 – 2 MeV	15% 85%	0.28 1.62
	(105K)	10/1954	1/1988			
		1992	7/1992			
	(105L)	7/1954	2/1968			
		1985	6/1988			
	(105P)	2/1954	8/1988			
	(105R)	12/1953	6/1964			
Plutonium Production	Plutonium Finishing Process: Plutonium enters the process as PuF_4 and is then fired into production pucks. Work is primarily conducted in glove boxes with predominant close anterior exposure to workers. Radiation levels at the beginning of the process are fairly constant while levels at end of process are closely related to production levels.					
	Plutonium Finishing Process (221-H, B-Line)	11/1954	1988	0.1-2 MeV 2-20 MeV	60% 40%	1.14 0.53
	Plutonium Finishing Process (221-F, B-Line)	10/1955	1988	0.1-2 MeV	100%	1.91
	Plutonium Storage			0.1-2 MeV 2-20 MeV	86% 14%	1.64 0.19
	Production Control Laboratory (772-F)	1955		0.1-2 MeV	100%	1.91
	Plutonium Fuel Facility (235-F)	1955		0.1-2 MeV	100%	1.91
Radionuclide Production and Calibration	Radiation characteristics are highly dependent upon the radionuclide being produced. SRS has produced several radionuclides including ^{238}Pu, ^{239}Pu, americium, curium, californium, etc. At the calibration facility radiological instruments are calibrated using a variety of sources with different neutron energy spectra.					
	Californium Production Facility (773-A)			0.1-2 MeV 2-20 MeV	50% 50%	0.95 0.66
	Calibration Facility (736-A)			0.1-2 MeV 2-20 MeV	83% 17%	1.58 0.23

5.5 Missed Dose

There is undoubtedly missed recorded dose for SRS workers. Analysis of the missed dose has been separated according to photon and neutron missed dose.

5.5.1 Photon Missed Dose

Missed photon dose for SRS workers would occur where: 1) there is no recorded dose because workers were not monitored or the dose is unavailable and 2) a zero dose is recorded for the respective SRS dosimeter systems for any dosimeter response less-than the minimum detection level (MDL). Methods to be considered when there is no recorded dose for a period within a working career have been examined by Watson et al (1994). Generally, estimates of the missed dose can be made using dose results for co-workers or using the recorded dose prior to and after the period of missed dose. However, these situations do require careful examination. The missed dose for dosimeter results less-than the MDL is particularly important for earlier years when MDLs were higher and dosimeter exchange was more frequent. NIOSH (2002) describes options to calculate the missed dose. One option is estimate a claimant favorable maximum potential missed dose where the Minimum Detection Level (MDL) is multiplied by the number of zero dose results providing an estimate of the maximum potential missed dose. The following sections describe the potential missed photon dose corrections according to facility/location, dosimeter type, year and energy range.

5.5.1.1 *Facility/Location*

The potential missed photon dose for the laboratory determined MDL and exchange frequency using NIOSH (2002) is presented in Table 5.5.1-1.

Table 5.5.1-1 Missed Photon Dose According to SRS Facility.

SRS Facility Type	Dosimeter	Period of Use	Exchange Frequency	MDL (mSv)[a]	Max. Annual Missed Dose (mSv)[b]
Fuel Fabrication, SRS Reactors, Processing Plants, Waste Handling	ORNL Two-Element film Dosimeter	Prior to 3/52	Weekly	0.4	20.8
	SRS Two Element film Dosimeter	After 3/52 through 9/30/1957	Weekly	0.4	20.8
		10/1/1957 through 11/8/1959	Biweekly	0.4	10.4
Above plus plutonium and nuclide production	SRS Multi-Element Film Dosimeter	11/9/1959 through 12/31/1964	Biweekly	0.4	10.4
		1/1/1965 through 12/31/1965	4-week	0.4	5.2
		1/1/1966 through 3/31/1970	Monthly	0.4	4.8
All SRS	SRS TLD	4/1/1970 through 6/30/1983	Monthly	0.15	1.8
	Panasonic TLD	7/1/1983 to 2003 (ongoing)	Monthly	0.05	0.6
	Panasonic TLND	1/1/1995 to 2003 (ongoing)	Monthly (n=12)	0.15	0.9

a. Estimated film dosimeter detection levels based on NIOSH (1993), NRC (1989), and Wilson et al (1990). TLD detection levels from Taylor et al (1995) and SRS (1993).
b. Maximum annual missed dose calculated from OCAS-IG-001 (NIOSH 2000).

5.5.1.2 *Dosimeter Type*

The MDLs for the respect SRS beta and photon dosimeters normally cited are based on laboratory irradiations. The actual MDLs are greater than these values because of additional uncertainty in actual field use. Reasonable MDLs are shown in Table 5.5.1-2 for most applications based on NIOSH (1993), NRC (1989), and Wilson et al (1990). The TLD detection levels from Taylor et al (1995) and SRS (1993).

Table 5.5.1-2 Missed Photon Dose According to Dosimeter Type

Dosimeter Type	Period of Use	MDL (mSv)	Exchange Frequency	Max. Annual Missed Dose (mSv)
Two Element Film	Prior to 3/52 through 9/30/1957	0.4	Weekly (n = 52)	20.8
	10/1/1957 through 11/8/1959	0.4	Biweekly (n=26)	10.4
Multi-Element Film	11/9/1959 through 12/31/1964	0.4	Biweekly (n=26)	10.4
	1/1/1965 through 12/31/1965	0.4	4-week (n=13)	5.2
	1/1/1966 through 3/31/1970	0.4	Monthly (n=12)	4.8
TLD	4/1/1970 through 6/30/1983	0.15	Monthly (n=12)	1.8
	7/1/1983 to 2003 (ongoing)	0.05	Monthly (n=12)	0.6
		0.05	Quarterly (n = 4)	0.2

5.5.1.3 *Year*

Analysis of the missed photon dose according to year (actually by period according to dosimeter type and exchange) may be needed to evaluate some claim information particularly if only annual dose data are available. Table 5.5.1-3 summarizes the potential missed dose.

Table 5.5.1-3 Missed Photon Dose Adjustments to Recorded Deep Dose

Period of Use	MDL (mSv)	Exchange Frequency	Max. Annual Missed Dose (mSv)
Prior to 3/52 through 9/30/1957	0.4	Weekly (n = 52)	20.8
10/1/1957 through 11/8/1959	0.4	Biweekly (n=26)	10.4
11/9/1959 through 12/31/1964	0.4	Biweekly (n=26)	10.4
1/1/1965 through 12/31/1965	0.4	4-week (n=13)	5.2
1/1/1966 through 3/31/1970	0.4	Monthly (n=12)	4.8
4/1/1970 through 6/30/1983	0.15	Monthly (n=12)	1.8
7/1/1983 to 2003 (ongoing)	0.05	Monthly (n=12)	0.6

5.5.1.4 *Energy Range*

An estimate of the missed photon dose by energy range is possible based on the type of facility and predominant radionuclides such as intermediate (>100 keV) energies for all facilities handling activation and fission product nuclides, primarily lower (<100 keV) energy photons for plutonium facilities and lower energy photons for uranium fuel fabrication facilities. The recorded dose from the dosimeter response does not typically

provide sufficient information to estimate discrete energy ranges. It is possible to examine the energy response characteristics of the respective multi-element dosimeters but this analysis does not recognize the substantial uncertainties present in the workplace associated with shielding, radiation scattering and mixed radiation fields.

5.5.2 Neutron Missed Dose

There is potential for significant missed dose among workers at the plutonium finishing facilities and the neutron source production area. There is also some potential for missed dose from work around the reactors and at the fuel fabrication facility. The approach used to calculate the neutron missed dose can be divided into two time periods. The first is before 1971 when neutron track emulsion type A (NTA) film was used and the second is when thermoluminescent dosimeters were used. Table 5.5.2-1 provides a summary of the reported limits of detection by the Savannah River Site (SRS).

Table 5.5.2-1 Summary of SRS Neutron Dosimeter Limits of Detection (LOD)

Dosimeter Type	Dates of Use		Exchange Frequency	Reported LOD (mrem)	Estimated LOD (mrem)	Max. Annual Missed Dose (mrem)
	Begin	End				
ORNL NTA	1951	8/3/1953	Weekly (n = 52)	~50[(a)]	~50	2600
SRS NTA	8/3/1953	7/13/1960	Weekly (n = 52)	30	~40[(b)]	2080
SRS NTA	7/14/1960	12/31/1970	Biweekly (n=26)	30	~40[(b)]	1040
SRS Hoy TLND	1/1/1971	12/31/1994	Monthly (n=12)	10[(c)]	20	240
Panasonic TLND	1/1/1995	Present	Monthly (n=12)	15	15	180

(a) Based on Wilson et al (1990)
(b) Correction for estimated 25% under response of NTA film for some facilities
(c) Minimum Recorded Dose not necessarily LOD

5.5.2.1 *Prior to 1971*

Due to the degraded energy spectra at most SRS facilities, the NTA film generally under responded due to the energy threshold of approximately 500 keV. The magnitude of the NTA film under response ranges from about 10% at the 773-A facility to over 90% around the reactors. According to Taylor et al. (1995), the SRS calibration method for NTA film might have accounted for some of this under response, however, until this can be further evaluated, the claimant favorable approach is to assume that it was not incorporated into the calibration methodology. As a result of the inability of NTA film to measure neutrons below 500 keV, the “true” limit of detection varies by energy spectra or facility. In the case of reactor operations, the limit of detection is virtually meaningless, while for the Californium Production facility (773-A), the reported limit of detection is probably reasonable however might be under reported by 10%.

Due to the uncertainty in whether an energy employee’s NTA badge would respond to the workplace neutron spectra, using a ratio to the measured photon dose is recommended as a claimant-favorable option to reconstruct an individual worker neutron dose. This is based on the fact that for routine neutron exposure, the neutron exposure is essentially always accompanied with measurable photon exposure. Table 5.5.2-2

provides a summary of the recommended neutron to photon ratios for the different process areas. As can be determined from the table, the recommended method to apply the ratio is as a log normal distribution using the geometric mean and geometric standard deviation. Caution, however, should be employed when using Table 5.5.2-2 and consideration should be given to an individual energy employee's neutron to photon ratio if there is sufficient individual data. For example if an individual has recorded neutron dose either prior to or after the estimation period that has a higher neutron to photon ratio than what is provided in the table, then the higher neutron to photon ratio should be used provided that the work assignments were similar for the two time periods. The values in Table 5.5.2-2 are intended to provide guidance when there is known technical limitations in the neutron dosimetry technology or no neutron monitoring data. The data were developed by evaluating multiple worker years, thus for a given time period, the "true" ratio might be more or less than the geometric mean. Due to potential for photon missed dose effecting the neutron to photon ratio, when applying the ratio methodology above, the photon dose should be the combination of both the measured photon dose (Dosimeter Dose) and the photon missed dose.

When overestimating dose reconstructions, the upper 95% ratio can be used as a claimant-favorable worst case ratio and the associated neutron uncertainty can be omitted. In underestimating dose reconstructions, the geometric mean can be used as a best case ratio.

Table 5.5.2-2 Summary of neutron to photon ratios by process

Process	Description / Buildings	Operations		Neutron to Photon Ratio		
		Begin	End	Geometric Mean (GM)	Geometric Standard Deviation (GSD)	Upper 95th%
Fuel Fabrication	Produced reactor fuel and target assemblies. Intermittent production mixed oxide fuel and plutonium – aluminum alloy targets.					
	(321-M)			(a)	(a)	(a)
Reactors	During Reactor Operation: Low level neutron exposure through shielding. In addition, some intermittent neutron exposure from mixed oxide fuel and targets.					
	(105C)	3/1955	1/1987	0.18	2.52	0.82
	(105K)	10/1954	1/1988			
		1992	7/1992			
	(105L)	7/1954	2/1968			
		1985	6/1988			
	(105P)	2/1954	8/1988			
	(105R)	12/1953	6/1964			
Plutonium Production	Plutonium Finishing Process: Plutonium enters the process as PuF_4 and is then fired into production pucks. Work is primarily conducted in glove boxes with predominant close anterior exposure to workers. Radiation levels at the beginning of the process are fairly constant while levels at end of process are closely related to production levels.					
	Plutonium Finishing Process (221-H, B-Line)	11/1954	1988	0.91	2.84	5.05
	Plutonium Finishing Process (221-F, B-Line)	10/1955	1988	0.36	2.52	1.65
	Plutonium Storage					
	Production Control Laboratory (772-F)	1955				

Process	Description / Buildings	Operations		Neutron to Photon Ratio		
		Begin	End	Geometric Mean (GM)	Geometric Standard Deviation (GSD)	Upper 95th%
	Plutonium Fuel Facility (235-F)	1955				
Radionuclide Production and Calibration	Radiation characteristics are highly dependent upon the radionuclide being produced. SRS has produced several radionuclides including ^{238}Pu, ^{239}Pu, americium, curium, californium, etc. At the calibration facility radiological instruments are calibrated using a variety of sources with different neutron energy spectra.					
	Californium Production Facility (773-A)			0.62	2.29	2.41
	Calibration Facility (736-A)			(b)	(b)	(b)

(a) During Pu-Al target production, workers with potential for neutron exposure wore NTA film, thus LOD values listed in table 5.5.2-1 should be used.
(b) Neutron to photon ratio for the calibration facility is not expected to be greater than the ratio observed in the californium production facility. As a claimant favorable approximation the Californium Production facility ratio can be used.

5.5.2.2 *Post 1971*

Since the introduction of thermoluminescent neutron dosimeters at the site, neutron exposures have been fairly well monitored and the recorded data is reasonably accurate. As a result the guidance on missed dose provided on the External dose Reconstruction Implementation Guideline (NIOSH 2002) should be followed in accordance with the LOD values presented in Table 5.5.2-1.

5.6 Organ Dose

Once the Hp(10) adjusted doses have been calculated for each year, these values are used according to NIOSH (2002) to calculate the organ dose distribution.

5.6.1 Organ Dose Conversion Factors

NIOSH (2002) describes the methodology used to calculate the organ dose distribution for the respective radiation types using identified exposure geometries. The selection of the worker orientation is important to the calculation of the organ dose. Examples of common exposure orientations are provided in the NIOSH Implementation Guide (2002), Table 4.2. Unfortunately, there is no definitive process to determine the exposure geometry for each worker. Table 5.6.1-1 lists proposed default options based on judgment of claimant favorable exposure geometries for long-term SRS workers.

Table 5.6.1-1 Default Exposure Geometries to calculate Organ Dose

Claim Status	Job Category (b)	Exposure Geometry	Percentage (c)
Likely non-compensable	All	AP	100%
Compensable - Workers	All	AP	50%
		ROT	50%
Compensable - Supervisors	All	AP	50%
		ISO	50%

Attachment E presents a discussion of the use of the various parameters presented in this section that will aid the dose reconstructionist in preparing dose reconstructions.

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ATTACHMENT A
DESCRIPTION OF FACILITIES AND PROCESSES

The primary operational areas and production facilities are discussed below.

A-1 **Reactors** – Five heavy-water reactors designated C, K, L, P, and R were constructed at SRS in the early 1950s, located in C Area, K Area, L Area, P Area, and R Area, respectively. The years of operation and radionuclides are shown in Table A-1.1 and A-1.2 below.

Table A-1.1 Reactors Years of Operation

Reactor	C	K	L	P	R
Years of Operation	1955 to1985	1954 to1988	1954 to1968 1985 to1988	1954 to1988	12/28/1953 to 06/1964

Table A-1.2 Radionuclides of Concern for All Reactors [(a)]

^{241}Am	^{60}Co	^{131}I	^{103}Ru
^{140}Ba	^{51}Cr	^{133}I	^{106}Ru
^{141}Ce	^{134}Cs	^{140}La	^{90}Sr / ^{90}Y
^{143}Ce	^{137}Cs	^{54}Mn	^{91}Sr
^{144}Ce	^{152}Eu	^{95}Nb	^{92}Sr
^{242}Cm	^{154}Eu	^{239}Np	^{234}U
^{244}Cm	^{59}Fe	^{238}Pu	^{65}Zn
^{58}Co	^{3}H [b]	^{239}Pu	^{95}Zr

(a) The composition and relative abundance are not known.
(b) Tritium is assumed to produce most of the personnel exposure because of the large quantities present.

A-2 **F-Area A-Line Facility** – The A-Line facility accepts dilute, depleted uranyl nitrate solution from the 221-F Facility and converts it into uranium trioxide powder. Dilute uranyl nitrate solution is sent to A-Line from the solvent extraction process in the 221-F Facility. The A-Line facility also has a dissolver to prepare a uranyl nitrate solution from the uranium trioxide product and nitric acid when it is required as a process feed stream back into the solvent extraction process in the 221-F Facility.

Table A-2 F Area A-Line Facility

Facility/Process	**Conversion of Dilute Depleted Uranyl Nitrate Solution (A-Line Feed) to** Uranium T**rioxide Powder** [(a) (b)]		
Years of Operation	1954-1987; 1993-1997		
Radionuclides Of Concern	**Absorption Types)**	**Activity Fraction**	**Significant to External Exposure**
^{239}Pu	M	5.559E-03	X
^{235}U	F[(c)]	8.022E-04	X
^{236}U	F[(c)]	8.179E-04	X
^{238}U	F[(c)]	7.440E-01	X
^{141}Ce	M	1.305E-02	X
^{144}Ce	M	6.079E-02	X
^{103}Ru	S[(d)]	5.969E-02	X
^{106}Ru	S[(d)]	6.111E-02	X
^{95}Nb	S[(d)]	5.418E-02	X

(a) Dosimetrically, the feed material for A-Line is ^{238}U and ^{239}Pu.
(b) For purpose of calculating internal dose, specific activity of the uranium is approximately 0.4 nCi/mg.
(c) Uranyl nitrate is type F; a solubility study on UO_3 powder at Hanford showed it to be mostly type F with a small component of type M; assume 80% F, 20% M.
(d) In the nitrate solution the ruthenium would be type F; after the oxidation process or after years of aging as contamination, the ruthenium would be type S. Type S will maximize the lung dose; type F maximizes the dose to all other tissues. Same statements apply to niobium except it is type M in the nitrate form.

A-3 221-F B-Line Facility – The F B-Line Facility, located in the 200-F Separations Area, is used to convert plutonium nitrate into plutonium metal or plutonium oxide and can recover plutonium from on-site and off-site scrap.

Table A-3 221-F B-Line Facility

Facility/Process	Conversion of plutonium nitrate into plutonium metal or plutonium oxide			
Years of Operation	1954-1989			
		6 % Mixture [(a) (b)]	**12 % Mixture** [(a) (c)]	
Radionuclides of Concern	**Absorption Types**	**Activity Fraction**	**Activity Fraction**	**Significant to External Exposure**
^{238}Pu	M-S[(d)]	4.800E-02	2.660E-01	X
^{239}Pu	M-S[(d)]	7.760E-01	4.660E-01	X
^{240}Pu	M-S[(d)]	1.760E-01	2.680E-01	X

(a) The percent refers to the mass percent of ^{240}Pu.
(b) The ratio of ^{241}Pu to alpha Pu is 8 to 1.
(c) The ratio of ^{241}Pu to alpha Pu is 23 to 1.
(d) Pu nitrate is type M; Pu oxides are type S.

A-4 221-F Canyon Facility – The 221-F Canyon Facility consists of two parallel canyons that constitute the process area. These canyons contain the high-activity and low-activity materials and are referred to as the hot and warm canyons respectively.

Table A-4.1 221-F Canyon Facility

Facility/Process	Head End Stream (Stream 1) Recovery of ^{239}Pu, ^{237}Np and ^{238}U [a]		
Years of Operation	11/1954-1989; 1993-2002		
Radionuclides of Concern	**Absorption Types**	**Activity Fraction**	**Significant to External Exposure [b]**
^{239}Pu	M	1.191E-03	X
^{144}Ce	M	4.765E-01	X
^{137}Cs	F	2.621E-02	X
^{106}Ru	F	1.588E-01	X
^{95}Zr	M	3.177E-01	X
^{90}Sr	F	1.959E-02	X

(a) The function of the 221-F Separation Facility is to isolate and purify plutonium, uranium, and neptunium from irradiated uranium. Fission products will also be present and, depending on the age of the material, the radionuclides of concern and the fraction of the activity of the materials will change. The majority of the internal dose from the Head End stream comes from plutonium, cerium, and ruthenium.
(b) The majority of the external dose from the Head End stream comes from cerium, zirconium, and ruthenium.

Table A-4.2 221-F Canyon Facility

Facility/Process	Second Uranium Cycle (Stream 2) [a]		
Years of Operation	11/1954-1989; 1993-2002		
Radionuclides of Concern	**Absorption Types**	**Activity Fraction**	**Significant to External Exposure [b]**
^{239}Pu	M	4.226E-03	
^{235}U	M	7.433E-04	
^{238}U	M	6.893E-01	X
^{144}Ce	M	5.633E-02	X
^{137}Cs	F	3.133E-03	X
^{106}Ru	F	5.662E-02	X
^{95}Zr	M	1.873E-01	X
^{90}Sr	F	2.332E-03	X

(a) The function of the 221-F Separation Facility is to isolate and purify plutonium, uranium, and neptunium from irradiated uranium. Fission products will also be present and, depending on the age of the material, the radionuclides of concern and the fraction of the activity of the materials will change. The majority of the internal dose from the second uranium cycle (Stream 2) comes from ^{238}U and ^{239}Pu as indicated above. It can be assumed this stream consists of depleted uranium with a specific activity of 0.4 nCi/mg and ^{239}Pu (La Bone, 1996).
(b) The majority of the external shallow dose from the second uranium cycle (Stream 2) comes from ^{238}U which feeds ^{234}mPa, and zirconium which emits both photons and betas.

Table A-4.3 221-F Canyon Facility

Facility/Process	Second Cycle Plutonium Stream (Stream 3) [a]		
Years of Operation	11/1954-1989; 1993-2002		
Radionuclides of Concern	**Absorption Types**	**Activity Fraction**	**Significant to External Exposure [b]**
^{238}Pu	M	2.045E-04	
^{239}Pu	M	6.075E-02	
^{241}Pu	M	9.194E-01	X
^{144}Ce	M	2.424E-03	X
^{106}Ru	F	9.204E-04	X
^{95}Zr	M	1.616E-02	X

(a) The function of the 221-F Separation Facility is to isolate and purify plutonium, uranium, and neptunium from irradiated uranium. Fission products will also be present, and, depending on the age of the material, the radionuclides of concern and the fraction of the activity of the materials will change.
(b) The majority of the external dose comes from ^{241}Pu and ^{95}Zr.

A-5 New Special Recovery Facility – Scrap plutonium from offsite sources and from FB-Line facility is processed through the New Special Recovery (NSR) Facility. The processed plutonium is conditioned so that it can be blended back into the plutonium nitrate feed stream to the FB-Line process.

Table A-5 New Special Recovery (NSR) Facility

Facility/Process	New Special Recovery (Pu Recovery)			
Years of Operation				
Radionuclides of Concern	**Absorption Types**	**Activity Fraction**		**Significant to External Exposure [b]**
^{241}Am	M			
^{239}Pu	M	To be determined		
^{240}Pu	M			
^{235}U	F			
^{236}U	F			
Examples of Feed Material for NSR Facility				
	Percent of Isotopic Content			
Material	^{239}Pu	^{240}Pu	^{241}Am	**Other**
				Fe, Ni, Cu, U
Impure Plutonium Oxide	87 to 94	6 to 12	0 to 1	-
Plutonium Metal	88 to 94	6 to 12	-	-
Unirradiated Reactor Cores	-	4 to 42	0 to 10	60 to 80 U
Scrap MgO Crucibles	88 to 94	6 to 12	0 to 1	< 90 MgO
Mixed Oxides and Metals	[c]	[c]	-	3 Mo
ZPPR Pu and U Oxide Plates	2 to 88	6 to 18	-	78 U, 2 Mo

(a) The majority of the internal dose is delivered by ^{239}Pu, ^{238}Pu, ^{242}Am, ^{235}U, and ^{236}U (assuming enriched uranium).
(b) The radionuclide that delivers the majority of the external dose is ^{241}Am. Some neutron dose will also be present due to spontaneous fission and an (alpha, n) reaction with fluoride, if present.
(c) Trace plutonium in uranium oxides

A-6 F-Area Outside Facilities – The F-Area Outside Facilities provide general support, principally to the processing of irradiated fuels and targets in Building 221-F. The term "Outside Facilities" is used to describe a wide variety of processes, utilities, and services that are ancillary to the primary 200-F Area operations.

Table A-6.1 F-Area Outside Facilities

Facility/Process Acid Recovery Unit (ARU) **Years of Operation** 1954-1989			
Radionuclides of Concern	**Absorption Types**	**Activity Fraction**	**Significant to External Exposure**
^{234}U	M	7.040E-11	
^{235}U	M	1.012E-09	
^{236}U	M	1.008E-09	
^{238}U	M	1.008E-07	
^{103}Ru	F	6.950E-02	X
^{106}Ru	F	1.023E-01	X
^{95}Zr	M	6.176E-01	X
^{95}Nb	M	2.107E-01	X

Table A-6.2 F Area Outside Facilities

Facility/Process **Water Handling System** **Years of Operation** **1954-1989**			
Radionuclides of Concern	**Absorption Types**	**Activity Fraction**	**Significant to External Exposure**
^{103}Ru	S	1.417E-01	X
^{106}Ru	S	4.815E-01	X
^{95}Zr	S	1.298E-01	X
^{95}Nb	S	2.470E-01	X

Table A-6.3 F Area Outside Facilities

Facility/Process	General Purpose Evaporators		
Years of Operation	1954-1989		
Radionuclides of Concern	**Absorption Types**	**Activity Fraction**	**Significant to External Exposure (b)**
^{238}Pu	S	9.626E-05	
^{239}Pu	S	2.903E-02	
^{241}Pu	S	3.913E-01	X
^{234}U	S	2.911E-08	
^{235}U	S	4.813E-07	
^{236}U	S	4.343E-07	
^{238}U	S	9.343E-05	X
^{103}Ru	S	4.852E-02	X
^{106}Ru	S	3.369E-01	X
^{95}Zr	S	9.743E-02	X
^{95}Nb	S	9.665E-02	X

(a) The radionuclide of concern for internal exposure for the GP Evaporators, Segregated Plutonium Solvent, and Laboratory Waste is plutonium. Type M (type F for ruthenium) should be used for nitrate form or from solvent extraction.
(b) The radionuclides of external dosimetric concern for the GP Evaporators, Segregated Plutonium Solvent, and Laboratory Waste are as indicated above. ^{238}U feeds ^{234}Pa with a high energy beta which is also of concern.

Table A-6.4 F Area Outside Facilities

Facility/Process	Segregated Plutonium Solvent		
Years of Operation	1954-1989		
Radionuclides of Concern	**Absorption Types**	**Activity Fraction**	**Significant to External Exposure (b)**
^{238}Pu	M	2.068E-04	
^{239}Pu	M	5.941E-02	
^{241}Pu	M	8.701E-01	X
^{103}Ru	F	2.418E-03	X
^{106}Ru	F	5.941E-02	X
^{95}Zr	M	5.941E-03	X
^{95}Nb	M	2.532E-03	X

(a) The radionuclide of concern for internal exposure at the GP Evaporators, Segregated Plutonium Solvent, and Laboratory Waste is plutonium.
(b) The radionuclides of external dosimetric concern for the GP Evaporators, Segregated Plutonium Solvent, and Laboratory Waste are as indicated above. ^{238}U feeds ^{234}Pa with a high energy beta which is also of concern.

Table A-6.5 F Area Outside Facilities Table

Facility/Process	Laboratory Waste		
Years of Operation	1954-1989		
Radionuclides of Concern	**Absorption Types[(a)]**	**Activity Fraction**	**Significant to External Exposure [(b)]**
^{238}Pu	S	6.150E-01	
^{239}Pu	S	1.921E-02	
^{241}Pu	S	3.456E-01	X
^{234}U	S	4.803E-08	
^{235}U	S	7.497E-07	
^{236}U	S	7.209E-07	
^{238}U	S	7.497E-05	X
^{141}Ce	M	1.369E-04	X
^{144}Ce	M	9.892E-03	X
^{134}Cs	F	1.822E-04	X
^{137}Cs	F	1.590E-03	X
^{103}Ru	S	1.369E-04	X
^{106}Ru	S	8.645E-04	X
^{95}Zr	M	6.150E-03	X
^{95}Nb	S	1.237E-03	X

(a) The radionuclide of concern for internal exposure at the GP Evaporators, Segregated Plutonium Solvent, and Laboratory Waste is plutonium. Various solubility classes are possible; those listed can be used as defaults. Nitrate forms would have been more soluble.
(b) The radionuclides of external dosimetric concern for the GP Evaporators, Segregated Plutonium Solvent, and Laboratory Waste are as indicated above. ^{238}U feeds ^{234}Pa with a high energy beta which is also of concern.

Table A-6.6 F Area Outside Facilities

Facility/Process	Segregated Uranium Solvent		
Years of Operation	1954-1989		
Radionuclides of Concern	**Absorption Types**	**Activity Fraction**	**Significant to External Exposure**
^{234}U	M	3.371E-07	
^{235}U	M	5.941E-06	
^{236}U	M	5.775E-06	
^{238}U	M	5.775E-04	X
^{103}Ru	F	1.491E-01	X
^{106}Ru	F	6.417E-01	X
^{95}Zr	M	1.491E-01	X
^{95}Nb	S	5.941E-02	X

A-7 $^{238}PuO_2$ Fuel Form Facility (PuFF) and the $^{238}PuO_2$ Experimental Facility (PEF) – The primary function of the PuFF facility is to produce encapsulated plutonium oxide fuel forms. The conversion of plutonium oxide powder into dense PuO_2 fuel forms is conducted in the PEF.

Table A-7 $^{238}PuO_2$ Fuel Form Facility (PuFF) and $^{238}PuO_2$ Experimental Facility (PEF)

Facility/Process	Production of encapsulated plutonium oxide fuel forms			
Years of Operation	1978 -??			
Radionuclides of Concern [a]	**Absorption Types**	**Activity Fraction**	**Specific Activity (Ci/g)**	**Significant to External Exposure**
^{241}Am	S[(b)]	3.067E-04	3.21	X
^{238}Pu	S	8.535E-01	1.66E+1	X
^{239}Pu	S	1.411E-01	6.13E-2	
^{240}Pu	S		2.27E-1	
^{241}Pu	S	4.089E-03	1.12E+2	
^{242}Pu	S		3.85E-3	
^{237}Np	M	5.111E-04	6.87E-4	X
^{232}Th	M	5.111E-04	1.11E-7	X

Particle Size	
Mass Median Diameter	
Ball-milled Powder	2 µm
Feed Powder	5 µm

(a) The majority of the dose comes from plutonium.
(b) Type M if mostly pure; type S if a contaminant in a plutonium matrix.

A-8 235-F Vaults – The 235-F facility receives plutonium and plutonium dioxide and fabricates it into a variety of items used in the nuclear program. There are three storage vaults located in the 235-F facility, the Plutonium Oxide Vault, Finished Product Vault, and Scrap Vault.

Table A-8 235-F Vaults

Facility/Process	Plutonium Oxide Vault; Finished Product Vault, Scrap Vault [(a)]		
Years of Operation			
Radionuclides of concern	**Absorption Types**	**Activity Fraction**	**Significant to External Exposure**
^{235}U	S		
^{238}Pu	S		
^{239}Pu	S		
^{237}Np	M		

(a) Detailed information on the composition of the radionuclides stored and found in these vaults is not available; therefore, they are considered to be pure.

A-9 772-F and 772-1F Production Control Laboratories – The production control laboratories provide analytical support for the 200-F and 200-H separations processes.

Table A-9 772-F and 772-1F Production Control Laboratories

Facility/Process 772-F & 772-1F Production Control Laboratories [(a)] **Years of Operation**			
Radionuclides of Concern	**Absorption Types**	**Activity Fraction**	**Significant to External Exposure**
Pu	M		
U	M		
Fission Products			

(a) Specific fission products or Pu and U information is not specified.

A-10 E-Area Solid Waste Disposal Facility (SWDF) – Solid and liquid low-level waste generated at the SRS and the Savannah River Laboratory (SRL) and some waste from off-site are stored at the SWDF.

Table A-10 E-Area Solid Waste Disposal Facility (SWDF)

Facility/Process Liquid and Solid Waste Disposal [(a)] **Years of Operation**			
Radionuclides of Concern	**Absorption Types**	**Activity Fraction**	**Significant to External Exposure**

(a) Due to the diverse nature of the radioactive waste, the radionuclide content of each type of waste and absorption types has not been defined. The radionuclides of highest concentration (Ci/vol) are tritium, cobalt, plutonium, and curium.

A-11 Plutonium Storage Facility (PSF) – The Plutonium Storage Facility (PSF) is located in the 200-F Separations Area and is used to receive, store, monitor, retrieve, and ship packaged plutonium, as needed.

Table A-11 Plutonium Storage Facility (PSF)

Facility/Process Receipt, Store, Monitor, Retrieve and Ship Plutonium **Years of Operation**				
Radionuclides of Concern	**Absorption Types**	**Activity Fraction [(a) (b) (d)] @ 6 % Pu Button**	**Activity Fraction [(a) (c) (d)] @12 % Pu Button**	**Significant to External Exposure**
^{238}Pu	S	4.800E-02	2.660E-01	
^{239}Pu	S	7.760E-01	4.660E-01	
^{240}Pu	S	1.760E-01	2.680E-01	

(a) The percent refers to the mass percent of 240Pu.
(b) The ratio of 241Pu to alpha Pu is 8.0 to 1.
(c) The ratio of 241Pu to alpha Pu is 23 to 1.
(d) The activity fractions of alpha-emitting plutonium (alpha-Pu) isotopes are those found in typical 6% or 12% plutonium buttons from the 221-F B Line.

A-12 F/H Effluent Treatment Facility (ETF) – The F/H Effluent Treatment Facility (ETF) consists of large impermeable storage basins with lift stations located in both F and H Areas and a treatment facility, Building 241-84H located in H Area. ETF is designed to remove hazardous chemical and radioactive contaminants from the 200-Area liquid effluent waste streams and concentrate them for disposal.

Table A-12 F/H Effluent Treatment Facility (ETF)

Facility/Process	Removal and Concentration of Waste from Liquid Waste Streams		
Years of Operation	1988- ??		
Radionuclides of Concern	**Absorption Types**	**Activity Fraction**	**Significant to External Exposure**
^{238}Pu	S (Y)[a]	2.855E-04	
^{239}Pu	S (Y)[a]	2.515E-06	
^{241}Pu	S (Y)[a]	8.401E-04	
^{144}Ce	M (W)[a]	4.690E-01	X
^{134}Cs	F	1.340E-02	X
^{137}Cs	F	2.723E-02	X
^{106}Ru	S	7.763E-02	X
^{95}Zr	M	2.382E-01	X
^{89}Sr	F	1.601E-01	X
^{90}Sr	F	1.335E-02	X

(a) Lung Solubility Classes were chosen to maximize the missed committed dose.

A-13 F & H Cooling Water and Retention Basins – The F and H Areas consist of chemical separation plants and appropriate support facilities for the production of purified radionuclides, principally uranium, plutonium, and tritium. The cooling water and retention basins are support facilities for the separations and waste management areas. Low-level radioactive liquid wastes from the separations areas are routed to the basins when activity is detected.

Table A-13 F & H Area - Cooling Water and Retention Basins

Facility/Process		**F- Area Retention Basins [a]**	**H - Area Retention Basins [a]**	
Years of Operation	1955-1989			
Radionuclides of concern	**Absorption Types**	**Activity Fraction**	**Activity Fraction**	**Significant to External Exposure [b]**
^{238}Pu	S	7.007E-03	1.403E-02	
^{239}Pu	S	2.803E-02	1.303E-02	
^{238}U	S	6.206E-02	9.018E-03	X
^{144}Ce	S	1.001E-03	1.403E-02	X
^{137}Cs	F	7.047E-01	7.335E-01	X
^{106}Ru	S	7.508E-02	2.405E-02	X
^{90}Sr	F	1.221E-01	1.924E-01	X

(a) The majority of the annual internal dose from the F-Area and H-Area Basins comes from plutonium and uranium.
(b) External exposure is for both F and H Area Retention Basins.

A-14 F & H Area Tank Farms – High-level liquid radioactive wastes from SRS are received and managed in large underground tanks in the waste tank farms in 241-F and 241-H. The two farms contain 51 large subsurface tanks and related facilities required for safe handling, processing, and temporary retention of liquid wastes.

Table A-14 F & H Area Tank Farms

Facility/Process		**F-Area Combined Tank Waste**	**H-Area Combined Tank Waste**	
Years of Operation		1955-1989		
Radionuclides of concern	**Absorption Types**	**Activity Fraction**	**Activity Fraction**	**Significant to External Exposure (b)**
^{241}Am	M	1.319E-04	7.346E-05	X
^{244}Cm	M	9.341E-04	6.121E-06	
^{238}Pu	M	2.198E-05	5.632E-03	X (c)
^{239}Pu	M	1.099E-04	-	
^{241}Pu	M	1.022E-02	5.693E-03	X
^{144}Ce	M	1.538E-01	2.632E-01	X
^{137}Cs	F	4.286E-01	3.489E-01	X
^{106}Ru	S	1.055E-02	2.142E-02	X
^{90}Sr	F	3.956E-01	3.550E-01	X

(a) When unknown or variable lung solubility classes were chosen to maximize the missed dose.
(b) External exposure is for both F and H Area Combined Tank Waste.
(c) For the F-Area Combined Tank Waste, ^{238}Pu is not a radionuclide of concern for external exposure.

A-15 Waste Certification Facility (WCF) – The Waste Certification Facility (WCF), Building 724-8E, located in the 643-E Solid Waste Disposal Facility, is used to certify and package drums of transuranic (TRU) waste. The functions of the WCF are to assay and certify the contents of TRU waste drums and to package, load, and ship the drums.

Table A-15 Waste Certification Facility (WCF)

Facility/Process Building 724-8E Certification and Packaging of TRU waste **Years of Operation**			
Radionuclides of concern (a)	**Absorption Types**	**Activity Fraction**	**Significant to External Exposure**

(a) Radionuclides are considered to be the same as for the Solid Waste Disposal Facility (643-E). The radionuclides in the solid waste disposal facility have not been identified. It is assumed the radionuclides are tritium, curium, plutonium, and activation products.

A-16 221-H B-Line Facility – The HB Line consists of three separate facilities; the

- Scrap Recovery Facility
- Neptunium Oxide Facility, and
- Plutonium Oxide Facility.

Table A-16.1 221-H B-Line Facility

Facility/Process Scrap Recovery ^{238}U, ^{239}U **Years of Operation**				
		Isotopic composition of ^{238}Pu scrap	**Isotopic composition of ^{239}Pu scrap**	
Radionuclides of concern	**Absorption Types**	**Activity Fraction**	**Activity Fraction**	**Significant to External Exposure (a)**
^{238}Pu	S	9.601E-01	-	X
^{239}Pu	S	1.200E-03	1.003E-01	X
^{241}Pu	S	3.870E-02	8.997E-01	X

(a) External exposure is for both ^{238}Pu and ^{239}Pu.

Table A-16.2 221-H B-Line Facility

Facility/Process Scrap Recovery ^{235}U **Years of Operation**			
	Isotopic composition of ^{235}U scrap		
Radionuclides of concern	**Absorption Types**	**Activity Fraction**	**Significant to External Exposure**
^{233}U	S	9.556E-01	X
^{235}U	S	4.020E-02	X
^{236}U	S	3.800E-03	X
^{238}U	S	4.000E-04	X

Table A-16.3 221-H B-Line Facility. $Np0_2$ and$Pu0_2$ Facility

Facility/Process **Years of Operation**		**Neptunium Oxide Facility**	**Plutonium Oxide Facility**	
		Isotopic composition of NpO_2 Material (a)	**Isotopic composition of PuO_2 Material**	
Radionuclides of concern	**Absorption Types**	**NpO_2 Activity Fraction**	**PuO_2 Activity Fraction**	**Significant to External Exposure (b)**
^{237}Np	M	1.000E+02	-	X
^{238}Pu	S	-	9.601E-01	X
^{239}Pu	S	-	8.992E-04	X
^{241}Pu	S	-	2.997E-02	X

(a) For purposes of dose calculations the NpO_2 is assumed to be 100% ^{237}Np.
(b) External exposure is for both facilities.

A-17 H-Canyon Facility – The H-Canyon building was designed for the separation and recovery of ^{239}Pu and ^{238}U from irradiated natural uranium by the Plutonium Uranium Extraction (PUREX) (Facility/Process) process. Over the years, other processes have been added. These operations include the processing of irradiated enriched uranium to recover ^{235}U, the processing of irradiated neptunium targets to separate and recover ^{238}Pu and ^{237}Np, and the processing of thorium to recover ^{233}U.

Table A-17.1 H-Canyon Facility

Facility/Process First Cycle Stream- Uranium Cycle [(a)]			
Years of Operation 1955-1991			
Radionuclides of concern	**Absorption Types**	**Activity Fraction**	**Significant to External Exposure**
^{238}Pu	M	5.719E-04	X
^{239}Pu	M	4.981E-06	
^{241}Pu	M	1.679E-03	X
^{237}Np	M	1.587E-07	
^{234}U	M	2.398E-06	
^{235}U	M	3.321E-08	
^{236}U	M	3.136E-07	
^{238}U	M	8.671E-10	
^{144}Ce	M	5.535E-01	X
^{137}Cs	F	3.321E-02	X
^{106}Ru	F	2.583E-02	X
^{95}Zr	M	2.214E-01	X
^{89}Sr	F	1.531E-01	X
^{90}Sr	F	1.070E-02	X

(a) The majority of the annual dose comes for the First Cycle Stream comes from Pu and Ce.

Table A-17.2 H-Canyon Facility

Facility/Process Second Uranium Cycle Stream [(a)]			
Years of Operation 1955-1991			
Radionuclides of concern	**Absorption Types**	**Activity Fraction**	**Significant to External Exposure**
^{238}Pu	M	9.016E-05	
^{239}Pu	M	7.889E-07	
^{241}Pu	M	2.630E-04	X
^{237}Np	M	5.071E-04	X
^{234}U	M	1.878E-01	X
^{235}U	M	2.630E-03	X
^{236}U	M	2.630E-02	X
^{238}U	M	6.950E-05	
^{144}Ce	M	4.320E-01	X
^{137}Cs	F	2.442E-02	X
^{106}Ru	F	2.066E-02	X
^{95}Zr	M	1.766E-01	X
^{89}Sr	F	1.202E-01	X
^{90}Sr	F	8.452E-03	X

(a) The majority of the annual dose from the Second Uranium Cycle Stream comes from uranium.

Table A-17.3 H-Canyon Facility

Facility/Process Neptunium Second Cycle [(a)] **Years of Operation** 1955-1991			
Radionuclides of concern	**Absorption Types[(b)]**	**Activity Fraction**	**Significant to External Exposure**
^{238}Pu	S	3.596E-03	X
^{239}Pu	S	3.083E-05	
^{241}Pu	S	1.028E-02	X
^{237}Np	M	4.624E-04	X
^{234}U	S	1.284E-06	
^{235}U	S	1.284E-06	
^{236}U	S	1.284E-06	
^{144}Ce	S	1.670E-01	X
^{137}Cs	F	1.002E-02	X
^{106}Ru	S	1.284E-02	X
^{95}Zr	M	6.936E-01	X
^{89}Sr	F	9.890E-02	X
^{90}Sr	F	3.339E-03	X

(a) The majority of the annual dose for the Second Neptunium Cycle Stream comes from plutonium and cerium.
(b) The plutonium would have been type M in the nitrate phase of the process and type S in the oxalate and oxide phases. Uranyl nitrate would have been type F; uranium trioxide: 80% F, 20% M.

Table A-17.4 H-Canyon Facility

Facility/Process Work Stream [(a)] **Years of Operation** 1955-1991			
Radionuclides of concern	**Absorption Types**	**Activity Fraction**	**Significant to External Exposure**
^{238}Pu	S	5.719E-04	X
^{239}Pu	S	4.981E-06	
^{241}Pu	S	1.679E-03	X
^{237}Np	M	1.476E-09	
^{234}U	S	2.214E-10	
^{235}U	S	3.690E-12	
^{236}U	S	2.952E-11	
^{238}U	S	7.380E-14	
^{144}Ce	S	5.535E-01	X
^{137}Cs	F	3.321E-02	X
^{106}Ru	S	2.583E-02	X
^{95}Zr	M	2.214E-01	X
^{89}Sr	F	1.531E-01	X
^{90}Sr	F	1.070E-02	X

(a) The majority of the annual dose for the Work Stream comes from plutonium and cerium.

A-18 211-H Area Outside Facilities – H-Area Outside Facilities include a number of processes that support the separations function of the 200-H Area.

Table A-18.1 H-Area Outside Facilities

Facility/Process A Line [(a)] **Years of Operation** 1955-1991			
Radionuclides of concern	**Absorption Types**	**Activity Fraction**	**Significant to External Exposure [(b)]**
^{234}U	F	8.055E-01	X
^{235}U	F	8.055E-03	X
^{236}U	F	1.859E-01	X
^{238}U	F	5.164E-04	X

(a) The radionuclide of concern for A-Line is the enriched uranium as indicated.
(b) The major contributor to external dose will be photons and x-rays from ^{234}U, ^{235}U, and ^{236}U.

Table A-18.2 H - Area Outside Facilities

Facility/Process Water Handling System **Years of Operation** 1955-1991			
Radionuclides of concern	**Absorption Types[(b)]**	**Activity Fraction**	**Significant to External Exposure [(a)]**
^{137}Cs	F	9.990E-04	X
^{106}Ru	S	7.992E-02	X
^{95}Zr	M	4.096E-01	X
^{95}Nb	S	5.095E-01	X

(a) The radionuclides for the other facilities are ^{95}Nb, ^{95}Zr, ^{103}Ru, ^{106}Ru, and ^{137}Cs. However, the mixture of radionuclides for the Water Handling System is considered typical of the other Outside Facilities: Acid Recovery Unit, General Purpose Evaporators, Segregated Solvent Facilities, Transfer Tanks, Sump Collection Tanks, and Recycle Sumps.
(b) Default type unless material is known to have been in the nitrate form, e.g, in the ARU, in which case Ru is type F, Zr is type F, and Nb is type M.

A-19 Receiving Basin for Off-site Fuel (RBOF) and Resin Regeneration Facility (RRF) – The RBOF facility handles casks and fuels of various shapes, sizes, and content. Casks of spent fuel are received and washed in the cask basin to remove any dirt accumulated during transit. The solubility classes in Table 19.1 are SR-1 for ^{131}I and SR-0 for gases or vapors that are inert and constitute a possible external exposure from submersion in cloud of gas and to internal exposure from gas within the respiratory tract. Class SR-1 refers to soluble or reactive gases or vapors with consideration given to retention in respiratory tract tissues, and to uptake to the systemic circulation, which may be less than the 100 % of the inhaled activity (ICRP 68, page9)

Table A-19.1 Receiving Basin for Off-site Fuel (RBOF)

Facility/Process Irradiated Fuel Receipt and Storage [(a)] **Years of Operation**			
Radionuclides of concern	**Absorption Types**	**Activity Fraction**	**Significant to External Exposure**
^{131}I	SR-1	1.020E-01	X
^{85}Kr-85	SR-0	8.900E-01	X
^{133}Xe	SR-0	5.600E-03	X

(a) The irradiated fuel contains uranium, fission products and activation products. Since the work is performed under water, the inert gases that escape the fuel bundles are a primary concern. The distribution above is for a Mark 22 Fuel Assembly cooled for 90 days.

Table A-19.2 Resin Regeneration Facility (RRF)

Facility/Process Spent Ion Exchange Resin Regeneration [(a)] **Years of Operation**			
Radionuclides of concern	**Absorption Types**	**Activity Fraction**	**Significant to External Exposure**
^{134}Cs	F	1.500E-01	X
^{137}Cs	F	1.500E-01	X
^{60}Co	S	7.000E-01	X

(a) The resins contain fission and activation products with the approximate distribution above. Irradiated target elements that are cut and cleaned at RRF contain tritium.

A-20 Tritium Facilities – The Tritium Facility includes Building 232-H, 233-H, 234-H, and 238-H. Irradiated lithium targets are processed in the tritium facilities to recover, purify, and package tritium.

Table A-20 Tritium Facilities [(a)]

Facility/Process Buildings 232-H, 233-H, 234-H, and 238-H Years of Operation 1952-Present			
Radionuclides of concern	Absorption Types	Activity Fraction	Significant to External Exposure
Elemental Tritium			
Tritiated water			
^{65}Zn	S		

(a) Elemental tritium and tritiated water are assumed to be the primary radionuclides of concern at the tritium facilities. In addition, the activation product ^{65}Zn is associated with the processing of lithium targets in building 232-H.

A-21 M-Area – Uranium Target Fabrication Facility, 313-M - The Uranium Target Fabrication Facility was designed and built to manufacture aluminum clad targets for irradiation in the SRS reactors.

Table A-21 Uranium Target Fabrication Facility, 313 M

Facility/Process	Uranium Target Fabrication [a]		
Years of Operation	1953-1989		
Radionuclides of concern	Absorption Types[b]	Activity Fraction	Significant to External Exposure
^{234}U	S	8.399E-02	X
^{235}U	S	1.450E-02	X
^{238}U	S	9.015E-01	X

(a) No information is available on the radionuclide distribution of the 313-M Facility. Therefore the distribution has been assumed. The majority of the annual dose from this mixture is delivered by ^{238}U.
(b) Based on solubility studies at similar facilities at Hanford, there may have been a type F component at about 10-20%. 100% type S should be assumed unless the urinalysis data show otherwise.

A-22 320-M, 322-M, and 341-M – Target extrusion was conducted in Building 320-M, and Building 322-M was a Chemical and Metallurgical Laboratory. The M-Area facilities processed uranium, lithium, and aluminum into fuel and target components for the nuclear reactors.

Table A-22 Uranium Processing Facilities 320-M, 322-M, and 341-M [a]

Facility/Process		Depleted Uranium [b]	Enriched Uranium [c]	
Years of Operation	1953-1989			
Radionuclides of concern	**Absorption Types[d]**	**Activity Fraction**	**Activity Fraction**	**Significant to External Exposure [e]**
^{234}U	S	8.399E-02	8.500E-01	X
^{235}U	S	1.450E-02	1.100E-02	X
^{236}U	S	-	1.380E-01	X
^{238}U	S	9.015E-01	1.000E-03	X

(a) No information is available on the processes or radionuclide distribution of the 320-M, 322-M, or 341-M Facilities, therefore the above distribution has been assumed.
(b) The majority of the annual internal dose from this mixture is delivered by ^{238}U.
(c) The majority of the annual internal dose from this mixture is delivered by ^{234}U.
(d) Based on solubility studies at similar facilities at Hanford, there may have been a type F component at about 10-20%. 100% type S should be assumed unless the urinalysis data show otherwise.
(e) The majority of the external shallow dose from this mixture is delivered by ^{238}U which decays to ^{234}Pa which emits a high energy beta. External exposure is for both types of uranium.

A-23 Fuel Fabrication Facility, 321-M – The Fuel Fabrication Facility was designed and built to manufacture aluminum clad fuel elements for irradiation in the SRS reactors.

Table A-23 Fuel Fabrication Facility, 321-M [(a)]

Facility/Process	Fuel Fabrication			
Years of Operation	1953-1989			
Radionuclide content of material being processed in 1996	Mass Fraction	Absorption Types[(b)]	Activity Fraction [(C)]	Significant to External Exposure
^{234}U	0.015	S	8.500E-01	X
^{235}U	0.580	S	1.100E-02	X
^{236}U	0.245	S	1.380E-01	X
^{238}U	0.160	S	1.000E-03	X
^{237}Np	1.0	-	-	X

(a) Plutonium dioxide was processed in 321-M in 1980 and Pu-Al Alloy work was conducted in the 1960s.
(b) Based on solubility studies at similar facilities at Hanford, there may have been a type F component at about 10-20%. 100% type S should be assumed unless the urinalysis data show otherwise.
(c) The majority of the dose from this mixture is delivered by ^{234}U.

A-24 S-Area- Defense Waste Processing Facility (DWPF) – Radioactive wastes are stored in existing H-Area waste tank facilities in the form of settled sludge, supernatant liquid, and salt cake. The insoluble solids (sludge) undergo aluminum dissolving and washing operations in preparation for the feed to DWPF.

Table A-24.1 S Area Defense Waste Processing Facility (DWPF)

Facility/Process	**Sludge Slurry Feed Stream (Stream 1)**			
Years of Operation	1996-???			
Radionuclides of concern	**Absorption Types**	**Activity Fraction [(a) (b)]**	**Solubility [(c)]**	**Significant to External Exposure**
^{244}Cm	M	1.682E-03	Unknown	X
^{238}Pu	S	2.326E-02	PuO_2, $PuO_2(NaTi_2O_5)_2$, $Na_2PuO_2(OH)_4$ (sol)	X
^{241}Pu	S	2.684E-02	Same as ^{238}Pu	X
^{144}Ce	S	1.557E-01	Unknown	X
^{137}Cs	F	2.326E-02	Cs_2O, CsOH (sol.)	X
^{106}Ru	S	3.578E-02	Na_2RuO_4,(sol.)	X
^{90}Sr	F	7.335E-01	$Sr(NaTi_2O_5)_2$, $Sr(OH)_2$(sol.), $SrCO_3$, SrO	X

(a) Specific activity 56 Ci/gal.
(b) The majority of the dose from this mixture is delivered by ^{90}Sr and ^{238}Pu.
(c) Chemical Form is insoluble unless otherwise indicated. Pu oxides are type S, but there is some indications that pure ^{238}Pu oxides are removed from the lung more rapidly than ^{239}Pu oxides.

Table A-24.2 S Area Defense waste Processing Facility (DWPF)

Facility/Process Precipitate-Slurry Feed Stream (Stream 201) **Years of Operation** 1996-???				
Radionuclides of concern	**Absorption Types**	**Activity Fraction [(a) b)]**	**Solubility [(c)]**	**Significant to External Exposure [(d)]**
^{244}Cm	M	2.537E-05	Unknown	X
^{238}Pu	S	3.171E-04	Same as sludge slurry	X
^{241}Pu	S	2.537E-04	Same as sludge slurry	X
^{144}Ce	S	1.839E-01		X
^{134}Cs	F	3.594E-03	CsO, CsOH (sol.), CsB(CH)	X
^{137}Cs	F	7.611E-01	Same as ^{134}Cs	X
^{106}Ru	S	4.228E-02		X
^{90}Sr	F	8.457E-03	Sr(NaTi O)$_2$, Sr(OH)(sol.), SrCO	X

(a) Specific activity is 37 Ci/gal.
(b) The majority of the internal dose from this mixture is delivered by ^{137}Cs and ^{144}Ce.
(c) Chemical Form is insoluble unless otherwise indicated.
(d) The majority of the external dose comes from ^{144}Ce, ^{137}Cs, ^{106}Ru, ^{90}Sr, ^{238}Pu, and ^{234}Pu.

A-25 Z Area – Waste tank supernate (low-level waste) is transferred from H Area to Z Area via pipeline. At Z Area, the salt solution (supernate) is mixed with cement, flyash, and blast furnace slag, and the resulting grout is pumped to vaults where it sets into a saltstone monolith.

Table A-25.1 Z Area

Facility/Process Waste Tank Supernate - Decontaminated Salt Solution **Years of Operation**				
Radionuclides of concern	**15 -Year Aged [(a)] (nCi/g)**	**Absorption Types**	**Activity Fraction**	**Significant to External Exposure [(b)]**
^{3}H	15.7			
^{90}Sr	0.8 (Including Y)	F	1.374E-2	X
Tc-99	53.0			
^{106}Ru	60 (including Rh)	M	5.888E-01	X
Sb-125	8.0			
^{137}Cs	9.4	F	3.925E-01	X
^{238}Pu	0.6	M	4.907E-03	X
^{239}Pu	0.007	M		
All TRU elements	0.9	M		

(a) The compositions are based on the tetraphenylborate and the sodium titanate ($NaTi_2O_5$) precipitation processes for removing ^{137}Cs and ^{90}Sr, respectively from the salt solution. Composition varies according to tanks.
(b) The majority of the external dose from by the salt solution is delivered by ^{238}Pu, ^{137}Cs, ^{90}Sr, and ^{106}Ru.

Table A-25.2 Z Area

Facility/Process Saltstone **Years of Operation**				
Radionuclides of concern	**15 -Year Aged [(a)] (nCi/g)**	**Absorption Types**	**Activity Fraction**	**Significant to External Exposure [(b)]**
^{3}H	7.0			
^{90}Sr	0.36 (double for Y)	S	1.401E-02	X
^{99}Tc	25.0			
^{106}Ru	26 (Double Rh)	M	5.602E-01	X
^{125}Sb	4.0			
^{137}Cs	4.4	F	4.202E-01	X
^{238}Pu	0.3	M	5.602E-03	X
^{239}Pu	0.003			
All TRU elements	0.4			

(a) The compositions are based on the tetra-phenylborate and the sodium titanate ($NaTi_2O_5$) precipitation processes for removing ^{137}Cs and ^{90}Sr, respectively from the salt solution. Composition varies according to tanks.
(b) In the saltstone, the majority of the external dose is delivered by ^{238}Pu, ^{137}Cs, ^{90}Sr, and ^{106}Ru.

A-26 A Area Building 773-A – Building 773-A has many processes including the following: Actinide Technology, Analytical Development, Defense Waste Processing Technology, Hydrogen Technology, Interim Waste Technology, Environmental Technology, Laboratory Services, Materials Technology.

Table A-26 A Area

Years of Operation 1954-1989
Additional documents are needed to review the facilities in the A Area for both internal and external radionuclides of concern. There are many "Sections" and "Technologies" in this area. No information is currently available

A-27 Buildings 735-A and 735-11A – Buildings 735-A and 735-11A house the Radiological and Environmental Science facilities.

A-28 Building 776-A Liquid Waste Handling Facility – Located in Building 776-A, this facility collects the aqueous waste from Building 773-A. Facilities in Building 776-A are provided to strain solids from the waste streams, collect waste in batch receiving tanks, sample and adjust the pH of the waste tank contents, and transfer the waste to tank trailers for disposal.

A-29 D-Area Heavy Water Production and Reprocessing – A heavy water production plant, in D-Area, began operation early in SRS history to concentrate heavy water from Savannah River to moderate and cool the site's reactors.

ATTACHMENT B
OCCUPATIONAL MEDICAL DOSE

B.0 SRS Medical X Ray

SRS conducted pre-employment and annual physical examinations as part of their occupational health program. These examinations typically included a chest x-ray. For some workers, and occupations, chest x-rays could be more frequent.

B.1 Organ Doses from Medical X-Rays

X-ray organ doses for occupational x-rays at SRS are estimated for all years from 1950 to present. The schedule for these exams for all SRS employees over this time period is shown in Table B-01 and B-02, along with the organ dose information. X-ray organ dose estimates were made for Type I equipment (used from 1950 to 1970), Type II equipment (used from 1971 to 7/1985), Type III equipment (used from 8/1985 to 5/1999), and Type IV equipment (used from 6/1999 to 2002). No collimation was assumed for Type I equipment used prior to 1970. For organs outside the chest cavity, other ICRP 34 organ dose conversion factors besides PA chest were used to ensure that the organ doses reflected their presence in the primary beam. For example, the dose to the ovaries for Type I equipment assumes they were in the primary beam, so the air kerma at the skin was multiplied by the dose conversion factor for ovaries in the abdomen.

Table B-01 Doses for Organs Identified in ICRP 34 (1982) Beam Quality for 3.5 mm Al HVL

Time Period	Frequency	SRS Employee	Air Kerma at Skin Entrance	Organ Doses per PA Chest [a] (rem)							
				Thyroid	Ovaries	Testes	Lungs	Breast	Embryo	Bone Marrow	WB
1950 - 1970 (Type I)[b]	Annual	All	0.108	0.107 [c]	0.032 [c]	0.0019 [c]	0.0610 (m) 0.0650 (f)	0.0098	0.028 [c]	0.0158(m) 0.0152(f)	0.0188 (m) 0.0174 (f)
1971-7/1985 (Type II)	Annual	All	0.044	0.0027	0.00014	<0.0000004	0.0249 (m) 0.0268 (f)	0.0040	0.00013	0.0064(m) 0.0062(f)	0.0076 (m) 0.0071 (f)
8/1985 - 5/1999 (Type III)	Annual	All	0.033	0.0020	0.00011	<0.0000003	0.0186 (m) 0.0201 (f)	0.0030	0.00010	0.0048(m) 0.0047(f)	0.0057 (m) 0.0053 (f)
6/1999 - 2003 (Type IV)	Annual	Over 50 years old	0.033	0.0020	0.00011	<0.0000003	0.0186 (m) 0.0201 (f)	0.0030	0.00010	0.0048(m) 0.0047(f)	0.0057 (m) 0.0053 (f)
	Biennial	40-49 years old									
	Every 3rd year	Under 40 years old									

[a] Organ doses for lateral chests can be estimated at 2.5 times the organ doses for PA chests (Ron Kathren and Vern Shockley)

[b] Doses from Type I equipment are estimated at 2.5 times the doses from Type II equipment (Ron Kathren and Vern Shockley)

[c] No collimation assumed prior to 1970. For organs not in the chest cavity, other ICRP dose conversion factors were used (abdomen, skull, and cervical spine).

Table B-02 Dose For IREP Organs Not Included in ICRP 34 (1982) Beam Quality for 3.5 mm Al HVL

Time Period	Frequency	SRS Employee	Organ Doses per PA Chest [a] (rem)											
			Air Kerma	Thymus	Liver/Gall Bladder	Eye/Brain	Esoph-agus	Stom-ach	Uterus	Urinary/ Bladder	Colon/ Rectum	Bone Surface	Skin[c]	Remain-der
1950-1970 (Type I)[b]	Annual	All	0.108	0.0650	0.032 [d]	0.0044	0.0650	0.0650	0.032 [d]	0.032 [d]	0.032 [d]	0.0650	0.151	0.0650
1971 - 7/1985 (Type II)	Annual	All	0.044	0.0268	0.00014	0.0027	0.0268	0.0268	0.00014	0.00014	0.00014	0.0268	0.062	0.0268
8/1985 - 5/199 (Type III)	Annual	Over 50 years old	0.033	0.0201	0.00011	0.0020	0.0201	0.0201	0.00011	0.00011	0.00011	0.0201	0.046	0.0201
	Biennial	40-49 years old												
	Every 3rd year	Under 40 years old												
1/1999 - 2003 (Type IV)	Biennial	Over 50 years old	0.033	0.0201	0.00011	0.0020	0.0201	0.0201	0.00011	0.00011	0.00011	0.0201	0.046	0.0201
	Every 3rd year	40-49 years old												
	Every 5th year	Under 40 years old												

[a] Organ doses for lateral chests can be estimated at 2.5 times the organ doses for PA chests (Ron Kathren and Vern Shockley)

[b] Doses from Type I equipment are estimated at 2.5 times the organ doses from Type II equipment (Ron Kathren and Vern Shockley)

[c] Entrance skin dose is entrance skin exposure calculated from air kerma, multiplied by a backscatter factor of 1.4 from NCRP 102, Table B-8.

[d] No collimation assumed prior to 1970. For organs not in the chest cavity, other ICRP 34 dose conversion factors were used (abdomen, skull, and cervical spine).

ATTACHMENT C
OCCUPATIONAL ENVIRONMENTAL DOSE

C.0 Occupational Environmental Dose

Unmonitored workers can be exposed to occupational doses internally from the resuspension of radioactive materials in soil and on-site radiation releases into the air, and externally from ambient radiation and releases of radioactive noble gases into air.

C.1 Internal Dose from On-Site Concentrations of Radionuclides

C.1.1. On-Site Releases to Air

If a worker spent significant time in areas for which radionuclide concentrations were calculated and the worker was not included in the internal dosimetry program, the data in Tables C-01 through C-15 may be used to estimate unmonitored intakes. These tables show average derived intake (assuming a 2,400 cubic meters per year ventilation Rate) at the 50th percentile and the geometric standard deviation, based on the 95th percentile source term for ^{3}H, ^{131}I, Pu, and U, in each SRS area by year. If the worker worked in multiple areas on the site, the site average annual intakes in Table C-16 should be used. Table C-17 shows the maximum annual intakes of these radionuclides at any SRS area by year. The decision to use these maximum air concentrations to estimate unmonitored intake should be made cautiously because the worker's intake could be significantly overestimated.

Table C-01 A Area Annual Intakes and geometric standard deviations, (Bq per year)

	Tritium		Iodine-131		Plutonium		Uranium	
Year	50th Percentile	GSD	50th Percentile	GSD	50th Percentile	GSD	50th Percentile	GSD
1955	2.73E+03	1.46	1.01E+02	1.79	9.54E-01	1.82	9.57E-02	1.82
1956	3.81E+04	1.26	2.26E+03	1.66	3.77E-03	1.82	4.39E-02	1.82
1957	1.25E+05	1.23	4.03E+02	1.89	5.04E-03	1.82	7.24E-04	1.82
1958	1.69E+05	1.21	2.60E+01	2.14	2.48E-03	1.26	4.77E-04	1.26
1959	6.21E+04	1.23	2.93E+02	1.86	2.80E-03	1.26	2.36E-03	1.26
1960	5.48E+04	1.24	1.33E+01	2.14	8.60E-03	1.26	6.60E-03	1.26
1961	5.01E+04	1.23	1.57E+02	1.81	1.34E-03	1.26	3.28E-03	1.26
1962	6.17E+04	1.26	4.87E+00	2.86	1.05E-03	1.26	2.32E-03	1.26
1963	5.94E+04	1.31	2.49E+00	3.19	3.57E-04	1.26	8.08E-03	1.26
1964	8.64E+04	1.30	1.12E+00	1.62	5.00E-04	1.26	1.40E-02	1.26
1965	4.22E+04	1.28	1.71E+00	1.36	1.39E-03	1.26	1.39E-02	1.26
1966	3.67E+04	1.28	2.92E+00	1.20	8.58E-04	1.26	5.47E-03	1.26
1967	3.86E+04	1.27	2.12E+00	1.57	9.09E-04	1.26	5.10E-03	1.26
1968	4.54E+04	1.27	2.33E+00	1.53	5.91E-04	1.26	5.95E-03	1.26
1969	2.77E+04	1.26	3.56E+00	1.50	1.22E-01	1.26	1.62E-02	1.26
1970	2.91E+04	1.25	3.13E+00	1.32	2.44E-03	1.26	4.28E-03	1.26
1971	3.84E+04	1.24	2.16E+00	1.22	2.42E-03	1.26	1.22E-03	1.26
1972	4.62E+04	1.23	2.68E-01	1.51	1.63E-03	1.26	1.90E-03	1.26
1973	3.16E+04	1.24	1.68E-01	1.28	1.86E-03	1.26	1.27E-03	1.26
1974	4.98E+04	1.51	1.65E-01	1.16	6.57E-04	1.26	2.33E-03	1.26
1975	2.77E+04	1.39	1.06E-02	1.17	1.99E-04	1.26	1.27E-03	1.26
1976	1.52E+04	1.26	1.34E-02	1.17	1.33E-03	1.26	1.33E-03	1.26
1977	2.13E+04	1.26	5.43E-03	1.25	4.13E-04	1.26	3.95E-04	1.26
1978	1.94E+04	1.28	5.66E-03	1.18	5.73E-04	1.26	7.88E-04	1.26
1979	1.83E+04	1.26	7.65E-03	1.32	1.58E-04	1.26	6.29E-04	1.26
1980	1.67E+04	1.25	2.51E-03	1.54	3.31E-04	1.26	1.04E-03	1.26
1981	2.07E+04	1.25	4.51E-03	1.44	6.12E-04	1.26	1.51E-03	1.26
1982	2.42E+04	1.24	1.02E-02	1.44	5.60E-04	1.26	2.42E-03	1.26
1983	3.48E+04	1.25	7.77E-03	1.34	2.92E-04	1.26	1.20E-03	1.26
1984	4.42E+04	1.25	2.50E-02	1.25	1.75E-04	1.26	5.62E-04	1.26
1985	3.77E+04	1.25	5.83E-03	1.43	1.03E-04	1.26	6.52E-04	1.26
1986	2.25E+04	1.23	2.45E-03	1.34	1.88E-04	1.26	3.80E-04	1.26
1987	3.20E+04	1.32	1.75E-03	2.04	1.87E-04	1.26	2.28E-03	1.26
1988	2.03E+04	1.23	8.78E-05	2.34	1.15E-04	1.26	3.87E-04	1.26
1989	1.51E+04	1.16	3.13E-05	1.15	1.57E-04	1.26	1.35E-03	1.26
1990	1.42E+04	1.16	1.04E-03	1.99	2.27E-04	1.82	3.59E-04	1.82
1991	1.08E+04	1.36	8.14E-04	1.99	5.09E-04	1.82	1.87E-04	1.82
1992	9.35E+03	1.19	2.39E-06	1.99	5.89E-05	1.82	1.20E-04	1.82
1993	1.03E+04	2.14	3.49E-04	1.99	1.32E-04	1.82	1.46E-04	1.82
1994	8.70E+03	1.71	1.28E-06	1.99	1.34E-04	1.82	1.81E-04	1.82
1995	5.32E+03	1.51	3.39E-04	1.99	1.13E-04	1.82	1.22E-04	1.82
1996	2.89E+03	1.58	8.00E-04	1.99	4.75E-05	1.82	1.33E-04	1.82
1997	3.24E+03	1.53	1.34E-04	1.99	3.94E-06	1.82	6.42E-06	1.82
1998	3.56E+03	1.48	1.68E-06	1.99	1.15E-05	1.82	3.76E-06	1.82
1999	2.27E+02	2.47	1.68E-06	1.99	4.07E-05	1.82	3.81E-06	1.82
2000	2.17E+02	2.27	1.11E-03	1.99	4.10E-05	1.82	8.12E-06	1.82
2001	2.26E+02	1.94	1.38E-03	1.99	8.30E-06	1.82	1.07E-05	1.82

Table C-02 C Reactor Area Annual Intakes and geometric standard deviations, (Bq per year)

	Tritium		Iodine-131		Plutonium		Uranium	
Year	50th Percentile	GSD	50th Percentile	GSD	50th Percentile	GSD	50th Percentile	GSD
1955	1.23E+04	1.42	2.33E+02	1.83	2.22E+00	1.82	2.16E-01	1.82
1956	1.39E+05	1.27	5.12E+03	1.67	8.81E-03	1.82	1.03E-01	1.82
1957	3.81E+05	1.23	9.29E+02	1.98	1.17E-02	1.82	1.69E-03	1.82
1958	5.93E+05	1.22	9.30E+01	2.74	5.78E-03	1.26	1.10E-03	1.26
1959	5.11E+05	1.23	7.12E+02	1.95	6.53E-03	1.26	5.52E-03	1.26
1960	5.86E+05	1.28	3.80E+01	2.43	2.02E-02	1.26	1.55E-02	1.26
1961	3.77E+05	1.26	3.81E+02	1.88	3.11E-03	1.26	7.64E-03	1.26
1962	4.42E+05	1.27	1.83E+01	3.08	2.40E-03	1.26	5.40E-03	1.26
1963	5.90E+05	1.32	1.60E+01	3.40	8.29E-04	1.26	1.88E-02	1.26
1964	8.46E+05	1.28	3.43E+00	2.04	1.13E-03	1.26	3.24E-02	1.26
1965	7.73E+05	1.32	4.70E+00	1.70	3.09E-03	1.26	3.26E-02	1.26
1966	6.04E+05	1.28	9.26E+00	1.80	1.91E-03	1.26	1.28E-02	1.26
1967	6.39E+05	1.28	7.38E+00	2.15	2.03E-03	1.26	1.19E-02	1.26
1968	8.05E+05	1.29	7.21E+00	2.05	1.33E-03	1.26	1.34E-02	1.26
1969	4.86E+05	1.30	9.99E+00	1.84	2.66E-01	1.26	3.69E-02	1.26
1970	4.53E+05	1.29	1.16E+01	2.14	5.39E-03	1.26	9.95E-03	1.26
1971	5.28E+05	1.27	7.00E+00	1.90	5.35E-03	1.26	2.78E-03	1.26
1972	5.62E+05	1.26	1.80E+00	2.72	3.59E-03	1.26	4.38E-03	1.26
1973	4.36E+05	1.27	4.88E-01	1.74	4.08E-03	1.26	2.92E-03	1.26
1974	5.08E+05	1.35	4.07E-01	1.31	1.47E-03	1.26	5.47E-03	1.26
1975	2.79E+05	1.31	2.50E-02	1.19	4.37E-04	1.26	2.98E-03	1.26
1976	2.58E+05	1.26	3.47E-02	1.41	2.94E-03	1.26	3.13E-03	1.26
1977	4.96E+05	1.29	2.10E-02	2.09	9.08E-04	1.26	9.17E-04	1.26
1978	3.95E+05	1.29	1.39E-02	1.30	1.26E-03	1.26	1.83E-03	1.26
1979	3.55E+05	1.33	2.05E-02	1.65	3.57E-04	1.26	1.46E-03	1.26
1980	3.80E+05	1.28	8.03E-03	2.05	7.33E-04	1.26	2.42E-03	1.26
1981	2.51E+05	1.27	2.11E-02	2.41	1.36E-03	1.26	3.48E-03	1.26
1982	3.88E+05	1.31	3.39E-02	2.04	1.29E-03	1.26	5.65E-03	1.26
1983	3.88E+05	1.29	2.64E-02	2.01	6.57E-04	1.26	2.80E-03	1.26
1984	7.16E+05	1.30	5.95E-02	1.29	4.01E-04	1.26	1.31E-03	1.26
1985	9.53E+05	1.28	1.38E-02	1.44	2.39E-04	1.26	1.52E-03	1.26
1986	2.38E+05	1.27	6.03E-03	1.45	4.16E-04	1.26	8.85E-04	1.26
1987	8.64E+04	1.30	4.22E-03	2.07	4.15E-04	1.26	5.35E-03	1.26
1988	6.31E+04	1.24	2.50E-04	2.50	2.66E-04	1.26	9.05E-04	1.26
1989	4.67E+04	1.13	7.34E-05	1.15	3.64E-04	1.26	3.17E-03	1.26
1990	4.22E+04	1.12	2.43E-03	1.99	5.16E-04	1.82	8.18E-04	1.82
1991	3.22E+04	1.26	1.91E-03	1.99	1.16E-03	1.82	4.25E-04	1.82
1992	2.69E+04	1.15	5.23E-06	1.99	1.34E-04	1.82	2.81E-04	1.82
1993	2.75E+04	2.73	7.95E-04	1.99	2.99E-04	1.82	3.42E-04	1.82
1994	2.10E+04	2.03	2.79E-06	1.99	2.99E-04	1.82	4.24E-04	1.82
1995	1.77E+04	1.58	7.80E-04	1.99	2.61E-04	1.82	2.87E-04	1.82
1996	9.34E+03	1.69	1.86E-03	1.99	1.08E-04	1.82	3.11E-04	1.82
1997	1.10E+04	1.60	2.92E-04	1.99	8.87E-06	1.82	1.44E-05	1.82
1998	9.77E+03	1.55	3.95E-06	1.99	2.59E-05	1.82	8.80E-06	1.82
1999	2.25E+03	2.00	3.95E-06	1.99	9.04E-05	1.82	8.88E-06	1.82
2000	4.99E+03	1.62	2.46E-03	1.99	9.14E-05	1.82	1.90E-05	1.82
2001	5.19E+03	1.52	3.06E-03	1.99	1.83E-05	1.82	2.51E-05	1.82

Table C-03 D Area Annual Intakes and geometric standard deviations, (Bq per year)

	Tritium		Iodine-131		Plutonium		Uranium	
Year	50th Percentile	GSD	50th Percentile	GSD	50th Percentile	GSD	50th Percentile	GSD
1955	2.12E+05	1.18	7.48E+01	1.80	6.96E-01	1.82	7.21E-02	1.82
1956	2.36E+05	1.08	1.70E+03	1.67	2.74E-03	1.82	3.18E-02	1.82
1957	3.09E+05	1.09	3.16E+02	1.91	3.66E-03	1.82	5.27E-04	1.82
1958	3.35E+05	1.11	2.29E+01	2.33	1.80E-03	1.26	3.53E-04	1.26
1959	5.40E+05	1.04	2.16E+02	1.88	2.04E-03	1.26	1.71E-03	1.26
1960	3.57E+05	1.13	1.07E+01	2.30	6.22E-03	1.26	4.77E-03	1.26
1961	2.92E+05	1.13	1.15E+02	1.84	9.84E-04	1.26	2.38E-03	1.26
1962	2.56E+05	1.17	6.02E+00	3.10	7.81E-04	1.26	1.69E-03	1.26
1963	2.60E+05	1.22	2.74E+00	3.30	2.61E-04	1.26	5.90E-03	1.26
1964	2.92E+05	1.22	1.06E+00	2.05	3.76E-04	1.26	1.03E-02	1.26
1965	2.58E+05	1.21	1.40E+00	1.64	1.06E-03	1.26	1.01E-02	1.26
1966	4.68E+05	1.13	2.22E+00	1.32	6.54E-04	1.26	3.97E-03	1.26
1967	4.70E+05	1.11	2.16E+00	2.09	6.92E-04	1.26	3.70E-03	1.26
1968	2.46E+05	1.17	2.26E+00	2.08	4.46E-04	1.26	4.49E-03	1.26
1969	2.98E+05	1.13	3.48E+00	2.02	9.50E-02	1.26	1.20E-02	1.26
1970	2.36E+05	1.12	2.54E+00	1.58	1.88E-03	1.26	3.13E-03	1.26
1971	2.31E+05	1.18	1.61E+00	1.31	1.87E-03	1.26	9.07E-04	1.26
1972	2.25E+05	1.18	2.29E-01	1.84	1.27E-03	1.26	1.40E-03	1.26
1973	1.97E+05	1.19	1.33E-01	1.51	1.45E-03	1.26	9.33E-04	1.26
1974	1.64E+05	1.31	1.20E-01	1.18	5.01E-04	1.26	1.69E-03	1.26
1975	1.72E+05	1.21	7.84E-03	1.23	1.54E-04	1.26	9.20E-04	1.26
1976	1.02E+05	1.20	9.76E-03	1.19	1.03E-03	1.26	9.65E-04	1.26
1977	1.06E+05	1.21	4.12E-03	1.37	3.19E-04	1.26	2.89E-04	1.26
1978	1.05E+05	1.20	4.19E-03	1.24	4.44E-04	1.26	5.75E-04	1.26
1979	8.35E+04	1.26	5.84E-03	1.47	1.19E-04	1.26	4.59E-04	1.26
1980	7.89E+04	1.21	2.11E-03	1.82	2.54E-04	1.26	7.58E-04	1.26
1981	8.01E+04	1.21	3.65E-03	1.68	4.70E-04	1.26	1.11E-03	1.26
1982	8.46E+04	1.23	8.26E-03	1.68	4.13E-04	1.26	1.76E-03	1.26
1983	9.64E+04	1.22	6.03E-03	1.50	2.21E-04	1.26	8.67E-04	1.26
1984	1.04E+05	1.23	1.87E-02	1.34	1.29E-04	1.26	4.10E-04	1.26
1985	1.01E+05	1.22	4.70E-03	1.64	7.59E-05	1.26	4.76E-04	1.26
1986	5.70E+04	1.21	2.08E-03	1.66	1.44E-04	1.26	2.77E-04	1.26
1987	3.76E+04	1.29	1.79E-03	2.38	1.43E-04	1.26	1.65E-03	1.26
1988	3.11E+04	1.24	1.20E-04	2.76	8.45E-05	1.26	2.81E-04	1.26
1989	2.59E+04	1.12	2.26E-05	1.15	1.14E-04	1.26	9.78E-04	1.26
1990	2.55E+04	1.11	7.50E-04	1.99	1.69E-04	1.82	2.68E-04	1.82
1991	2.32E+04	1.16	5.88E-04	1.99	3.80E-04	1.82	1.39E-04	1.82
1992	1.08E+04	1.17	1.87E-06	1.99	4.41E-05	1.82	8.71E-05	1.82
1993	2.16E+04	2.37	2.61E-04	1.99	9.84E-05	1.82	1.06E-04	1.82
1994	1.49E+04	1.84	9.97E-07	1.99	1.02E-04	1.82	1.32E-04	1.82
1995	1.21E+04	1.35	2.51E-04	1.99	8.33E-05	1.82	8.84E-05	1.82
1996	1.01E+04	1.34	5.83E-04	1.99	3.56E-05	1.82	9.59E-05	1.82
1997	1.09E+04	1.28	1.04E-04	1.99	2.98E-06	1.82	4.88E-06	1.82
1998	1.23E+04	1.22	1.22E-06	1.99	8.72E-06	1.82	2.72E-06	1.82
1999	5.82E+03	1.26	1.22E-06	1.99	3.12E-05	1.82	2.77E-06	1.82
2000	5.80E+03	1.21	8.55E-04	1.99	3.13E-05	1.82	5.90E-06	1.82
2001	5.78E+03	1.16	1.05E-03	1.99	6.42E-06	1.82	7.72E-06	1.82

Table C-04 E Area Annual Intakes and geometric standard deviations, (Bq per year)

	Tritium		Iodine-131		Plutonium		Uranium	
Year	50th Percentile	GSD	50th Percentile	GSD	50th Percentile	GSD	50th Percentile	GSD
1955	1.64E+04	1.50	7.40E+02	1.79	6.86E+00	1.82	7.20E-01	1.82
1956	2.61E+05	1.27	1.70E+04	1.66	2.70E-02	1.82	3.13E-01	1.82
1957	9.25E+05	1.23	3.14E+03	1.88	3.61E-02	1.82	5.19E-03	1.82
1958	1.25E+06	1.21	1.93E+02	2.03	1.77E-02	1.26	3.50E-03	1.26
1959	4.36E+05	1.23	2.10E+03	1.84	2.01E-02	1.26	1.68E-02	1.26
1960	3.54E+05	1.25	8.96E+01	2.05	6.11E-02	1.26	4.69E-02	1.26
1961	3.43E+05	1.23	1.11E+03	1.80	9.72E-03	1.26	2.35E-02	1.26
1962	4.02E+05	1.27	1.85E+01	2.28	7.76E-03	1.26	1.66E-02	1.26
1963	3.97E+05	1.27	8.40E+00	2.80	2.57E-03	1.26	5.83E-02	1.26
1964	5.27E+05	1.32	6.99E+00	1.33	3.75E-03	1.26	1.02E-01	1.26
1965	2.21E+05	1.29	1.16E+01	1.24	1.07E-02	1.26	9.89E-02	1.26
1966	2.07E+05	1.29	2.04E+01	1.15	6.56E-03	1.26	3.90E-02	1.26
1967	2.06E+05	1.27	1.33E+01	1.30	6.94E-03	1.26	3.64E-02	1.26
1968	2.68E+05	1.27	1.51E+01	1.30	4.46E-03	1.26	4.49E-02	1.26
1969	1.71E+05	1.24	2.31E+01	1.27	9.61E-01	1.26	1.20E-01	1.26
1970	1.76E+05	1.22	2.15E+01	1.22	1.89E-02	1.26	3.08E-02	1.26
1971	2.39E+05	1.22	1.51E+01	1.19	1.88E-02	1.26	9.02E-03	1.26
1972	3.08E+05	1.22	1.78E+00	1.34	1.28E-02	1.26	1.39E-02	1.26
1973	2.02E+05	1.22	1.16E+00	1.20	1.46E-02	1.26	9.24E-03	1.26
1974	3.44E+05	1.54	1.17E+00	1.16	5.02E-03	1.26	1.66E-02	1.26
1975	1.81E+05	1.40	7.50E-02	1.16	1.56E-03	1.26	9.04E-03	1.26
1976	9.28E+04	1.23	9.48E-02	1.16	1.04E-02	1.26	9.48E-03	1.26
1977	1.30E+05	1.24	3.78E-02	1.20	3.22E-03	1.26	2.86E-03	1.26
1978	1.21E+05	1.27	4.00E-02	1.16	4.49E-03	1.26	5.67E-03	1.26
1979	1.11E+05	1.23	5.24E-02	1.21	1.19E-03	1.26	4.53E-03	1.26
1980	9.95E+04	1.22	1.62E-02	1.30	2.56E-03	1.26	7.47E-03	1.26
1981	1.41E+05	1.23	3.00E-02	1.28	4.73E-03	1.26	1.10E-02	1.26
1982	1.58E+05	1.22	6.73E-02	1.27	4.09E-03	1.26	1.73E-02	1.26
1983	2.37E+05	1.23	5.29E-02	1.23	2.22E-03	1.26	8.52E-03	1.26
1984	2.99E+05	1.23	1.74E-01	1.19	1.28E-03	1.26	4.05E-03	1.26
1985	2.37E+05	1.23	3.85E-02	1.25	7.51E-04	1.26	4.69E-03	1.26
1986	1.59E+05	1.21	1.65E-02	1.22	1.45E-03	1.26	2.74E-03	1.26
1987	2.40E+05	1.31	9.16E-03	1.57	1.44E-03	1.26	1.62E-02	1.26
1988	1.40E+05	1.22	4.07E-04	1.81	8.36E-04	1.26	2.77E-03	1.26
1989	9.42E+04	1.20	2.22E-04	1.15	1.13E-03	1.26	9.62E-03	1.26
1990	9.17E+04	1.20	7.37E-03	1.99	1.68E-03	1.82	2.67E-03	1.82
1991	6.96E+04	1.44	5.78E-03	1.99	3.78E-03	1.82	1.39E-03	1.82
1992	6.38E+04	1.21	1.89E-05	1.99	4.40E-04	1.82	8.56E-04	1.82
1993	7.31E+04	1.66	2.59E-03	1.99	9.81E-04	1.82	1.04E-03	1.82
1994	6.42E+04	1.51	1.01E-05	1.99	1.02E-03	1.82	1.30E-03	1.82
1995	3.96E+04	1.44	2.49E-03	1.99	8.25E-04	1.82	8.68E-04	1.82
1996	2.08E+04	1.47	5.75E-03	1.99	3.54E-04	1.82	9.42E-04	1.82
1997	2.44E+04	1.44	1.05E-03	1.99	2.98E-05	1.82	4.89E-05	1.82
1998	2.72E+04	1.43	1.20E-05	1.99	8.73E-05	1.82	2.68E-05	1.82
1999	1.27E+03	1.92	1.20E-05	1.99	3.14E-04	1.82	2.73E-05	1.82
2000	1.06E+03	1.89	8.61E-03	1.99	3.14E-04	1.82	5.80E-05	1.82
2001	1.22E+03	1.68	1.06E-02	1.99	6.48E-05	1.82	7.59E-05	1.82

Table C-05 F Area Annual Intakes and geometric standard deviations, (Bq per year)

	Tritium		Iodine-131		Plutonium		Uranium	
Year	50th Percentile	GSD	50th Percentile	GSD	50th Percentile	GSD	50th Percentile	GSD
1955	4.50E+04	1.61	2.06E+03	1.79	2.46E+01	1.82	1.46E+00	1.82
1956	1.02E+06	1.26	3.52E+04	1.66	1.02E-01	1.82	1.22E+00	1.82
1957	1.80E+06	1.21	2.11E+03	1.88	1.35E-01	1.82	1.94E-02	1.82
1958	2.37E+06	1.20	9.54E+01	2.09	6.74E-02	1.26	1.01E-02	1.26
1959	2.23E+05	1.24	7.58E+03	1.84	7.48E-02	1.26	6.73E-02	1.26
1960	1.88E+05	1.27	3.57E+02	2.00	2.47E-01	1.26	1.89E-01	1.26
1961	1.75E+05	1.25	4.50E+03	1.80	3.26E-02	1.26	8.75E-02	1.26
1962	2.06E+05	1.29	5.07E+01	1.56	2.06E-02	1.26	6.20E-02	1.26
1963	2.05E+05	1.32	1.73E+01	1.95	9.06E-03	1.26	2.05E-01	1.26
1964	2.78E+05	1.35	2.69E+01	1.18	7.87E-03	1.26	3.39E-01	1.26
1965	1.34E+05	1.33	4.62E+01	1.18	1.41E-02	1.26	3.99E-01	1.26
1966	1.19E+05	1.32	8.21E+01	1.12	9.71E-03	1.26	1.53E-01	1.26
1967	1.23E+05	1.31	5.16E+01	1.16	1.08E-02	1.26	1.43E-01	1.26
1968	1.55E+05	1.30	5.91E+01	1.19	8.64E-03	1.26	8.70E-02	1.26
1969	9.95E+04	1.26	9.07E+01	1.18	4.65E-01	1.26	3.07E-01	1.26
1970	1.02E+05	1.24	8.60E+01	1.17	1.98E-02	1.26	1.10E-01	1.26
1971	1.36E+05	1.24	6.09E+01	1.16	1.84E-02	1.26	2.31E-02	1.26
1972	1.67E+05	1.23	6.88E+00	1.21	9.56E-03	1.26	4.24E-02	1.26
1973	1.13E+05	1.23	4.63E+00	1.17	8.88E-03	1.26	2.91E-02	1.26
1974	1.82E+05	1.52	4.72E+00	1.15	7.66E-03	1.26	6.73E-02	1.26
1975	9.80E+04	1.39	2.99E-01	1.16	1.10E-03	1.26	3.67E-02	1.26
1976	5.37E+04	1.25	3.85E-01	1.16	9.30E-03	1.26	3.81E-02	1.26
1977	7.89E+04	1.25	1.50E-01	1.17	2.60E-03	1.26	9.85E-03	1.26
1978	7.12E+04	1.28	1.62E-01	1.16	3.22E-03	1.26	2.02E-02	1.26
1979	6.58E+04	1.25	2.08E-01	1.17	2.41E-03	1.26	1.63E-02	1.26
1980	6.06E+04	1.24	6.24E-02	1.19	2.91E-03	1.26	2.71E-02	1.26
1981	7.55E+04	1.24	1.18E-01	1.19	5.60E-03	1.26	3.49E-02	1.26
1982	8.86E+04	1.24	2.65E-01	1.18	1.27E-02	1.26	6.68E-02	1.26
1983	1.27E+05	1.24	2.11E-01	1.17	3.98E-03	1.26	3.35E-02	1.26
1984	1.65E+05	1.24	7.01E-01	1.16	3.63E-03	1.26	1.43E-02	1.26
1985	1.43E+05	1.25	1.52E-01	1.17	2.43E-03	1.26	1.69E-02	1.26
1986	8.26E+04	1.23	6.60E-02	1.17	1.65E-03	1.26	9.88E-03	1.26
1987	1.16E+05	1.32	3.26E-02	1.26	1.80E-03	1.26	6.55E-02	1.26
1988	7.04E+04	1.23	1.33E-03	1.36	2.72E-03	1.26	1.07E-02	1.26
1989	4.88E+04	1.20	9.05E-04	1.15	3.96E-03	1.26	3.84E-02	1.26
1990	4.70E+04	1.19	3.00E-02	1.99	4.15E-03	1.82	6.58E-03	1.82
1991	3.58E+04	1.41	2.36E-02	1.99	9.31E-03	1.82	3.42E-03	1.82
1992	3.23E+04	1.21	9.02E-06	1.99	1.01E-03	1.82	3.37E-03	1.82
1993	3.66E+04	1.90	6.40E-03	1.99	2.31E-03	1.82	4.09E-03	1.82
1994	3.15E+04	1.61	4.82E-06	1.99	1.62E-03	1.82	5.00E-03	1.82
1995	1.94E+04	1.47	7.31E-03	1.99	2.58E-03	1.82	3.52E-03	1.82
1996	1.03E+04	1.53	2.08E-02	1.99	8.42E-04	1.82	3.84E-03	1.82
1997	1.19E+04	1.49	5.23E-04	1.99	5.58E-05	1.82	7.91E-05	1.82
1998	1.32E+04	1.46	4.88E-05	1.99	1.55E-04	1.82	1.06E-04	1.82
1999	7.18E+02	2.25	4.88E-05	1.99	3.77E-04	1.82	1.02E-04	1.82
2000	6.84E+02	2.12	9.26E-03	1.99	4.46E-04	1.82	2.23E-04	1.82
2001	7.49E+02	1.84	1.41E-02	1.99	5.22E-05	1.82	3.08E-04	1.82

Table C-06 H Area Annual Intakes and geometric standard deviations, (Bq per year)

	Tritium		Iodine-131		Plutonium		Uranium	
Year	50th Percentile	GSD	50th Percentile	GSD	50th Percentile	GSD	50th Percentile	GSD
1955	3.04E+04	1.28	1.51E+03	1.79	7.08E+00	1.82	2.17E+00	1.82
1956	1.46E+05	1.25	5.03E+04	1.66	2.05E-02	1.82	1.92E-01	1.82
1957	2.88E+06	1.24	1.50E+04	1.87	2.94E-02	1.82	4.23E-03	1.82
1958	3.93E+06	1.20	9.41E+02	1.98	1.30E-02	1.26	6.67E-03	1.26
1959	2.12E+06	1.22	2.05E+03	1.84	1.70E-02	1.26	7.84E-03	1.26
1960	1.68E+06	1.22	3.74E+01	2.13	2.60E-02	1.26	2.09E-02	1.26
1961	1.72E+06	1.20	4.46E+02	1.81	1.29E-02	1.26	1.97E-02	1.26
1962	2.01E+06	1.24	1.29E+01	2.81	1.73E-02	1.26	1.38E-02	1.26
1963	1.97E+06	1.22	7.78E+00	3.22	2.83E-03	1.26	6.41E-02	1.26
1964	2.54E+06	1.28	3.05E+00	1.58	1.10E-02	1.26	1.36E-01	1.26
1965	1.00E+06	1.22	4.76E+00	1.33	4.19E-02	1.26	4.35E-02	1.26
1966	1.01E+06	1.23	8.37E+00	1.19	2.45E-02	1.26	2.34E-02	1.26
1967	9.22E+05	1.22	5.71E+00	1.50	2.52E-02	1.26	2.04E-02	1.26
1968	1.27E+06	1.22	6.43E+00	1.46	1.40E-02	1.26	1.41E-01	1.26
1969	8.03E+05	1.21	9.74E+00	1.43	4.82E+00	1.26	2.80E-01	1.26
1970	8.58E+05	1.19	8.72E+00	1.29	8.10E-02	1.26	3.25E-02	1.26
1971	1.10E+06	1.21	6.23E+00	1.21	8.23E-02	1.26	2.12E-02	1.26
1972	1.47E+06	1.20	7.70E-01	1.46	5.97E-02	1.26	2.37E-02	1.26
1973	1.00E+06	1.19	4.93E-01	1.25	7.10E-02	1.26	1.46E-02	1.26
1974	1.68E+06	1.53	5.02E-01	1.16	1.84E-02	1.26	6.87E-03	1.26
1975	8.69E+05	1.39	3.69E-02	1.17	7.37E-03	1.26	3.73E-03	1.26
1976	4.69E+05	1.20	3.81E-02	1.17	4.66E-02	1.26	4.40E-03	1.26
1977	5.77E+05	1.21	1.75E-02	1.23	1.48E-02	1.26	3.41E-03	1.26
1978	5.58E+05	1.24	1.67E-02	1.17	2.11E-02	1.26	5.96E-03	1.26
1979	5.06E+05	1.20	2.40E-02	1.29	3.61E-03	1.26	4.54E-03	1.26
1980	4.50E+05	1.19	7.95E-03	1.47	1.07E-02	1.26	7.20E-03	1.26
1981	7.13E+05	1.21	1.27E-02	1.41	1.94E-02	1.26	1.70E-02	1.26
1982	7.41E+05	1.20	2.88E-02	1.41	6.65E-03	1.26	1.14E-02	1.26
1983	1.16E+06	1.22	2.22E-02	1.32	7.37E-03	1.26	4.88E-03	1.26
1984	1.41E+06	1.21	7.20E-02	1.24	2.57E-03	1.26	4.42E-03	1.26
1985	1.05E+06	1.21	1.68E-02	1.39	1.11E-03	1.26	4.70E-03	1.26
1986	8.77E+05	1.19	6.98E-03	1.30	6.06E-03	1.26	2.70E-03	1.26
1987	1.28E+06	1.30	4.76E-03	1.98	5.80E-03	1.26	7.14E-03	1.26
1988	7.17E+05	1.21	2.28E-04	2.27	1.20E-03	1.26	1.86E-03	1.26
1989	4.38E+05	1.21	8.65E-05	1.15	1.27E-03	1.26	4.80E-03	1.26
1990	4.33E+05	1.20	2.86E-03	1.99	4.15E-03	1.82	6.58E-03	1.82
1991	3.28E+05	1.46	2.24E-03	1.99	9.31E-03	1.82	3.42E-03	1.82
1992	3.06E+05	1.21	9.47E-05	1.99	1.18E-03	1.82	4.92E-04	1.82
1993	3.54E+05	1.45	6.38E-03	1.99	2.56E-03	1.82	6.06E-04	1.82
1994	3.16E+05	1.42	5.06E-05	1.99	3.66E-03	1.82	8.59E-04	1.82
1995	1.95E+05	1.41	4.61E-03	1.99	1.33E-03	1.82	3.70E-04	1.82
1996	1.01E+05	1.42	5.67E-03	1.99	9.11E-04	1.82	3.71E-04	1.82
1997	1.21E+05	1.41	5.27E-03	1.99	9.60E-05	1.82	1.74E-04	1.82
1998	1.35E+05	1.41	4.64E-06	1.99	2.92E-04	1.82	1.39E-05	1.82
1999	5.55E+03	1.52	4.64E-06	1.99	1.29E-03	1.82	2.29E-05	1.82
2000	4.08E+03	1.53	3.66E-02	1.99	1.20E-03	1.82	4.05E-05	1.82
2001	4.98E+03	1.47	4.14E-02	1.99	2.98E-04	1.82	3.19E-05	1.82

Table C-07 S Area Annual Intakes and geometric standard deviations, (Bq per year)

	Tritium		Iodine-131		Plutonium		Uranium	
Year	50th Percentile	GSD	50th Percentile	GSD	50th Percentile	GSD	50th Percentile	GSD
1955	1.31E+04	1.35	6.23E+02	1.79	3.68E+00	1.82	8.18E-01	1.82
1956	1.03E+05	1.27	1.90E+04	1.66	1.22E-02	1.82	1.28E-01	1.82
1957	1.08E+06	1.24	5.24E+03	1.87	1.70E-02	1.82	2.44E-03	1.82
1958	1.47E+06	1.21	3.30E+02	2.00	7.90E-03	1.26	2.80E-03	1.26
1959	7.35E+05	1.23	1.09E+03	1.85	9.65E-03	1.26	6.11E-03	1.26
1960	5.83E+05	1.23	3.23E+01	2.13	2.15E-02	1.26	1.68E-02	1.26
1961	5.82E+05	1.21	3.83E+02	1.81	6.06E-03	1.26	1.12E-02	1.26
1962	6.79E+05	1.26	1.12E+01	2.81	6.96E-03	1.26	7.88E-03	1.26
1963	6.68E+05	1.24	6.87E+00	3.23	1.43E-03	1.26	3.24E-02	1.26
1964	8.72E+05	1.30	2.64E+00	1.56	4.17E-03	1.26	6.37E-02	1.26
1965	3.37E+05	1.25	4.10E+00	1.32	1.51E-02	1.26	3.52E-02	1.26
1966	3.28E+05	1.26	7.14E+00	1.18	8.88E-03	1.26	1.58E-02	1.26
1967	3.16E+05	1.24	4.94E+00	1.48	9.19E-03	1.26	1.43E-02	1.26
1968	4.22E+05	1.25	5.51E+00	1.45	5.24E-03	1.26	5.28E-02	1.26
1969	2.67E+05	1.23	8.40E+00	1.42	1.67E+00	1.26	1.12E-01	1.26
1970	2.78E+05	1.20	7.53E+00	1.28	2.87E-02	1.26	1.67E-02	1.26
1971	3.78E+05	1.21	5.28E+00	1.21	2.91E-02	1.26	8.42E-03	1.26
1972	5.01E+05	1.21	6.45E-01	1.45	2.09E-02	1.26	1.03E-02	1.26
1973	3.28E+05	1.20	4.12E-01	1.25	2.48E-02	1.26	6.48E-03	1.26
1974	5.71E+05	1.55	4.13E-01	1.16	6.70E-03	1.26	5.78E-03	1.26
1975	2.97E+05	1.40	2.80E-02	1.17	2.58E-03	1.26	3.14E-03	1.26
1976	1.48E+05	1.22	3.27E-02	1.17	1.64E-02	1.26	3.45E-03	1.26
1977	1.97E+05	1.22	1.38E-02	1.23	5.20E-03	1.26	1.67E-03	1.26
1978	1.88E+05	1.26	1.40E-02	1.17	7.40E-03	1.26	3.07E-03	1.26
1979	1.70E+05	1.21	1.93E-02	1.30	1.36E-03	1.26	2.38E-03	1.26
1980	1.51E+05	1.20	6.32E-03	1.49	3.80E-03	1.26	3.85E-03	1.26
1981	2.33E+05	1.23	1.09E-02	1.40	6.94E-03	1.26	7.60E-03	1.26
1982	2.52E+05	1.21	2.46E-02	1.41	2.93E-03	1.26	7.32E-03	1.26
1983	3.91E+05	1.22	1.89E-02	1.32	2.73E-03	1.26	3.38E-03	1.26
1984	4.82E+05	1.22	6.13E-02	1.24	1.07E-03	1.26	2.24E-03	1.26
1985	3.59E+05	1.22	1.42E-02	1.40	5.04E-04	1.26	2.47E-03	1.26
1986	2.72E+05	1.20	5.93E-03	1.30	2.16E-03	1.26	1.43E-03	1.26
1987	4.20E+05	1.31	4.10E-03	1.98	2.08E-03	1.26	5.78E-03	1.26
1988	2.40E+05	1.22	1.93E-04	2.25	5.51E-04	1.26	1.18E-03	1.26
1989	1.57E+05	1.20	7.57E-05	1.15	6.38E-04	1.26	3.60E-03	1.26
1990	1.54E+05	1.20	2.51E-03	1.99	1.63E-03	1.82	2.59E-03	1.82
1991	1.16E+05	1.45	1.97E-03	1.99	3.66E-03	1.82	1.34E-03	1.82
1992	1.08E+05	1.21	3.29E-05	1.99	4.55E-04	1.82	3.40E-04	1.82
1993	1.24E+05	1.52	2.51E-03	1.99	9.92E-04	1.82	4.16E-04	1.82
1994	1.10E+05	1.45	1.76E-05	1.99	1.33E-03	1.82	5.49E-04	1.82
1995	6.81E+04	1.42	1.95E-03	1.99	5.88E-04	1.82	3.06E-04	1.82
1996	3.56E+04	1.43	3.00E-03	1.99	3.55E-04	1.82	3.22E-04	1.82
1997	4.21E+04	1.42	1.83E-03	1.99	3.57E-05	1.82	6.35E-05	1.82
1998	4.70E+04	1.41	4.07E-06	1.99	1.08E-04	1.82	1.02E-05	1.82
1999	2.00E+03	1.67	4.07E-06	1.99	4.59E-04	1.82	1.30E-05	1.82
2000	1.52E+03	1.68	1.30E-02	1.99	4.32E-04	1.82	2.52E-05	1.82
2001	1.82E+03	1.55	1.49E-02	1.99	1.05E-04	1.82	2.66E-05	1.82

Table C-08 Z Area Annual Intakes and geometric standard deviations, (Bq per year)

	Tritium		Iodine-131		Plutonium		Uranium	
Year	50th Percentile	GSD	50th Percentile	GSD	50th Percentile	GSD	50th Percentile	GSD
1955	8.30E+03	1.40	3.78E+02	1.79	2.50E+00	1.82	4.69E-01	1.82
1956	7.88E+04	1.27	1.09E+04	1.66	8.79E-03	1.82	9.53E-02	1.82
1957	6.14E+05	1.24	2.84E+03	1.88	1.20E-02	1.82	1.73E-03	1.82
1958	8.36E+05	1.21	1.80E+02	2.02	5.71E-03	1.26	1.72E-03	1.26
1959	4.00E+05	1.24	7.51E+02	1.85	6.80E-03	1.26	4.75E-03	1.26
1960	3.20E+05	1.25	2.62E+01	2.16	1.69E-02	1.26	1.31E-02	1.26
1961	3.15E+05	1.23	3.05E+02	1.82	3.95E-03	1.26	7.92E-03	1.26
1962	3.69E+05	1.27	1.02E+01	2.90	4.17E-03	1.26	5.57E-03	1.26
1963	3.62E+05	1.27	6.86E+00	3.29	9.64E-04	1.26	2.18E-02	1.26
1964	4.77E+05	1.32	2.17E+00	1.62	2.40E-03	1.26	4.15E-02	1.26
1965	1.88E+05	1.27	3.30E+00	1.35	8.35E-03	1.26	2.76E-02	1.26
1966	1.79E+05	1.28	5.68E+00	1.19	4.95E-03	1.26	1.18E-02	1.26
1967	1.77E+05	1.27	4.02E+00	1.53	5.14E-03	1.26	1.08E-02	1.26
1968	2.31E+05	1.26	4.44E+00	1.49	2.98E-03	1.26	3.01E-02	1.26
1969	1.47E+05	1.23	6.79E+00	1.46	9.03E-01	1.26	6.63E-02	1.26
1970	1.52E+05	1.21	6.02E+00	1.29	1.58E-02	1.26	1.13E-02	1.26
1971	2.10E+05	1.22	4.19E+00	1.22	1.59E-02	1.26	5.00E-03	1.26
1972	2.75E+05	1.21	5.16E-01	1.47	1.14E-02	1.26	6.42E-03	1.26
1973	1.79E+05	1.21	3.27E-01	1.27	1.34E-02	1.26	4.10E-03	1.26
1974	3.12E+05	1.55	3.25E-01	1.16	3.74E-03	1.26	4.57E-03	1.26
1975	1.64E+05	1.41	2.16E-02	1.17	1.40E-03	1.26	2.49E-03	1.26
1976	8.05E+04	1.23	2.60E-02	1.17	8.97E-03	1.26	2.68E-03	1.26
1977	1.10E+05	1.23	1.08E-02	1.24	2.83E-03	1.26	1.11E-03	1.26
1978	1.04E+05	1.26	1.11E-02	1.18	4.02E-03	1.26	2.08E-03	1.26
1979	9.49E+04	1.22	1.53E-02	1.33	7.78E-04	1.26	1.63E-03	1.26
1980	8.40E+04	1.21	5.06E-03	1.55	2.09E-03	1.26	2.65E-03	1.26
1981	1.26E+05	1.23	8.75E-03	1.43	3.82E-03	1.26	4.83E-03	1.26
1982	1.38E+05	1.21	1.99E-02	1.45	1.84E-03	1.26	5.38E-03	1.26
1983	2.13E+05	1.23	1.52E-02	1.34	1.54E-03	1.26	2.54E-03	1.26
1984	2.64E+05	1.22	4.89E-02	1.26	6.49E-04	1.26	1.51E-03	1.26
1985	2.00E+05	1.23	1.15E-02	1.44	3.22E-04	1.26	1.69E-03	1.26
1986	1.44E+05	1.21	4.75E-03	1.33	1.19E-03	1.26	9.80E-04	1.26
1987	2.24E+05	1.31	3.51E-03	2.08	1.15E-03	1.26	4.54E-03	1.26
1988	1.30E+05	1.22	1.68E-04	2.33	3.54E-04	1.26	8.66E-04	1.26
1989	8.81E+04	1.20	6.04E-05	1.15	4.28E-04	1.26	2.77E-03	1.26
1990	8.55E+04	1.20	2.00E-03	1.99	9.62E-04	1.82	1.53E-03	1.82
1991	6.44E+04	1.44	1.57E-03	1.99	2.16E-03	1.82	7.92E-04	1.82
1992	5.91E+04	1.21	1.77E-05	1.99	2.65E-04	1.82	2.56E-04	1.82
1993	6.80E+04	1.60	1.48E-03	1.99	5.80E-04	1.82	3.12E-04	1.82
1994	6.01E+04	1.48	9.48E-06	1.99	7.48E-04	1.82	4.04E-04	1.82
1995	3.71E+04	1.43	1.20E-03	1.99	3.70E-04	1.82	2.41E-04	1.82
1996	1.94E+04	1.45	2.06E-03	1.99	2.08E-04	1.82	2.57E-04	1.82
1997	2.28E+04	1.43	9.88E-04	1.99	2.03E-05	1.82	3.56E-05	1.82
1998	2.55E+04	1.42	3.25E-06	1.99	6.09E-05	1.82	7.78E-06	1.82
1999	1.13E+03	1.82	3.25E-06	1.99	2.53E-04	1.82	9.20E-06	1.82
2000	8.77E+02	1.83	7.14E-03	1.99	2.40E-04	1.82	1.84E-05	1.82
2001	1.03E+03	1.64	8.26E-03	1.99	5.70E-05	1.82	2.10E-05	1.82

Table C-09 Forest Service Headquarters [(a)] Calculated Air Concentration (Bq per m^3)

	Tritium		Iodine-131		Plutonium		Uranium	
Year	50th Percentile	GSD	50th Percentile	GSD	50th Percentile	GSD	50th Percentile	GSD
1955	2.36E+03	1.49	9.55E+01	1.80	8.19E-01	1.82	9.94E-02	1.82
1956	3.09E+04	1.28	2.34E+03	1.66	3.15E-03	1.82	3.61E-02	1.82
1957	1.30E+05	1.24	4.88E+02	1.88	4.23E-03	1.82	6.08E-04	1.82
1958	1.75E+05	1.22	3.16E+01	2.11	2.06E-03	1.26	4.47E-04	1.26
1959	7.08E+04	1.26	2.51E+02	1.86	2.36E-03	1.26	1.91E-03	1.26
1960	5.95E+04	1.29	1.10E+01	2.18	6.95E-03	1.26	5.34E-03	1.26
1961	5.63E+04	1.27	1.27E+02	1.82	1.19E-03	1.26	2.76E-03	1.26
1962	6.70E+04	1.33	5.06E+00	2.99	1.01E-03	1.26	1.95E-03	1.26
1963	6.45E+04	1.36	2.99E+00	3.30	3.09E-04	1.26	6.99E-03	1.26
1964	8.84E+04	1.40	9.45E-01	1.71	5.15E-04	1.26	1.24E-02	1.26
1965	4.06E+04	1.36	1.39E+00	1.39	1.56E-03	1.26	1.13E-02	1.26
1966	3.60E+04	1.36	2.36E+00	1.21	9.50E-04	1.26	4.50E-03	1.26
1967	3.76E+04	1.34	1.75E+00	1.62	9.99E-04	1.26	4.18E-03	1.26
1968	4.59E+04	1.32	1.91E+00	1.58	6.21E-04	1.26	6.26E-03	1.26
1969	3.02E+04	1.26	2.92E+00	1.54	1.51E-01	1.26	1.58E-02	1.26
1970	3.16E+04	1.24	2.53E+00	1.33	2.83E-03	1.26	3.69E-03	1.26
1971	4.25E+04	1.24	1.74E+00	1.23	2.84E-03	1.26	1.19E-03	1.26
1972	5.26E+04	1.23	2.18E-01	1.53	1.96E-03	1.26	1.75E-03	1.26
1973	3.54E+04	1.23	1.36E-01	1.30	2.27E-03	1.26	1.15E-03	1.26
1974	5.79E+04	1.53	1.33E-01	1.16	7.24E-04	1.26	1.88E-03	1.26
1975	3.19E+04	1.40	8.62E-03	1.18	2.40E-04	1.26	1.03E-03	1.26
1976	1.67E+04	1.25	1.08E-02	1.17	1.58E-03	1.26	1.08E-03	1.26
1977	2.31E+04	1.25	4.40E-03	1.25	4.93E-04	1.26	3.45E-04	1.26
1978	2.13E+04	1.28	4.57E-03	1.18	6.91E-04	1.26	6.78E-04	1.26
1979	2.00E+04	1.25	6.31E-03	1.38	1.65E-04	1.26	5.39E-04	1.26
1980	1.79E+04	1.24	2.14E-03	1.64	3.81E-04	1.26	8.87E-04	1.26
1981	2.36E+04	1.25	3.71E-03	1.49	7.01E-04	1.26	1.37E-03	1.26
1982	2.71E+04	1.23	8.46E-03	1.51	5.11E-04	1.26	2.00E-03	1.26
1983	3.99E+04	1.24	6.38E-03	1.39	3.12E-04	1.26	9.80E-04	1.26
1984	5.02E+04	1.24	2.04E-02	1.29	1.65E-04	1.26	4.86E-04	1.26
1985	4.08E+04	1.25	4.90E-03	1.51	9.28E-05	1.26	5.59E-04	1.26
1986	2.58E+04	1.23	2.00E-03	1.37	2.16E-04	1.26	3.25E-04	1.26
1987	3.86E+04	1.32	1.63E-03	2.20	2.12E-04	1.26	1.85E-03	1.26
1988	2.40E+04	1.24	8.01E-05	2.45	1.03E-04	1.26	3.21E-04	1.26
1989	1.75E+04	1.19	2.52E-05	1.15	1.36E-04	1.26	1.10E-03	1.26
1990	1.63E+04	1.19	8.35E-04	1.99	2.24E-04	1.82	3.55E-04	1.82
1991	1.21E+04	1.41	6.55E-04	1.99	5.02E-04	1.82	1.84E-04	1.82
1992	1.08E+04	1.21	2.96E-06	1.99	5.93E-05	1.82	9.85E-05	1.82
1993	1.22E+04	1.97	3.44E-04	1.99	1.32E-04	1.82	1.20E-04	1.82
1994	1.05E+04	1.64	1.58E-06	1.99	1.46E-04	1.82	1.50E-04	1.82
1995	6.43E+03	1.49	3.16E-04	1.99	1.03E-04	1.82	9.87E-05	1.82
1996	3.44E+03	1.55	6.84E-04	1.99	4.74E-05	1.82	1.07E-04	1.82
1997	3.91E+03	1.50	1.65E-04	1.99	4.17E-06	1.82	7.00E-06	1.82
1998	4.34E+03	1.46	1.35E-06	1.99	1.23E-05	1.82	3.07E-06	1.82
1999	2.29E+02	2.41	1.35E-06	1.99	4.66E-05	1.82	3.21E-06	1.82
2000	1.97E+02	2.33	1.29E-03	1.99	4.57E-05	1.82	6.75E-06	1.82
2001	2.15E+02	1.98	1.55E-03	1.99	9.91E-06	1.82	8.62E-06	1.82

(a) Values are for plant perimeter.

Table C-10 Central Shops Annual Intakes and geometric standard deviations, (Bq per year)

	Tritium		Iodine-131		Plutonium		Uranium	
Year	50th Percentile	GSD	50th Percentile	GSD	50th Percentile	GSD	50th Percentile	GSD
1955	4.45E+04	1.21	1.49E+02	1.81	1.31E+00	1.82	1.51E-01	1.82
1956	9.48E+04	1.19	3.57E+03	1.67	5.06E-03	1.82	5.83E-02	1.82
1957	2.49E+05	1.21	7.28E+02	1.90	6.79E-03	1.82	9.76E-04	1.82
1958	3.27E+05	1.20	5.29E+01	2.31	3.32E-03	1.26	6.97E-04	1.26
1959	2.37E+05	1.17	4.06E+02	1.89	3.79E-03	1.26	3.09E-03	1.26
1960	1.97E+05	1.25	1.92E+01	2.29	1.13E-02	1.26	8.65E-03	1.26
1961	1.60E+05	1.24	2.08E+02	1.84	1.88E-03	1.26	4.42E-03	1.26
1962	1.73E+05	1.29	1.05E+01	3.09	1.57E-03	1.26	3.13E-03	1.26
1963	1.82E+05	1.35	6.03E+00	3.34	4.91E-04	1.26	1.11E-02	1.26
1964	2.42E+05	1.37	1.83E+00	1.99	7.86E-04	1.26	1.96E-02	1.26
1965	1.76E+05	1.34	2.47E+00	1.59	2.34E-03	1.26	1.82E-02	1.26
1966	1.93E+05	1.26	4.21E+00	1.43	1.43E-03	1.26	7.26E-03	1.26
1967	2.00E+05	1.25	3.49E+00	1.95	1.50E-03	1.26	6.75E-03	1.26
1968	1.83E+05	1.29	3.58E+00	1.87	9.44E-04	1.26	9.51E-03	1.26
1969	1.44E+05	1.23	5.35E+00	1.77	2.22E-01	1.26	2.44E-02	1.26
1970	1.32E+05	1.22	4.79E+00	1.68	4.22E-03	1.26	5.88E-03	1.26
1971	1.52E+05	1.24	3.10E+00	1.47	4.21E-03	1.26	1.84E-03	1.26
1972	1.65E+05	1.23	5.26E-01	2.19	2.90E-03	1.26	2.74E-03	1.26
1973	1.26E+05	1.24	2.39E-01	1.49	3.35E-03	1.26	1.81E-03	1.26
1974	1.54E+05	1.42	2.18E-01	1.20	1.09E-03	1.26	3.05E-03	1.26
1975	1.01E+05	1.31	1.40E-02	1.19	3.55E-04	1.26	1.66E-03	1.26
1976	6.63E+04	1.24	1.80E-02	1.24	2.34E-03	1.26	1.75E-03	1.26
1977	9.84E+04	1.27	8.32E-03	1.59	7.29E-04	1.26	5.48E-04	1.26
1978	8.56E+04	1.27	7.53E-03	1.23	1.02E-03	1.26	1.08E-03	1.26
1979	7.71E+04	1.29	1.07E-02	1.50	2.51E-04	1.26	8.60E-04	1.26
1980	7.49E+04	1.26	3.91E-03	1.85	5.68E-04	1.26	1.42E-03	1.26
1981	6.95E+04	1.25	7.63E-03	1.92	1.05E-03	1.26	2.15E-03	1.26
1982	8.84E+04	1.27	1.56E-02	1.75	8.02E-04	1.26	3.22E-03	1.26
1983	1.07E+05	1.26	1.17E-02	1.64	4.73E-04	1.26	1.58E-03	1.26
1984	1.53E+05	1.27	3.36E-02	1.33	2.57E-04	1.26	7.73E-04	1.26
1985	1.64E+05	1.27	8.24E-03	1.59	1.46E-04	1.26	8.91E-04	1.26
1986	6.73E+04	1.25	3.42E-03	1.49	3.23E-04	1.26	5.19E-04	1.26
1987	6.34E+04	1.32	2.97E-03	2.31	3.17E-04	1.26	2.99E-03	1.26
1988	4.28E+04	1.25	1.82E-04	2.67	1.62E-04	1.26	5.16E-04	1.26
1989	3.19E+04	1.18	4.08E-05	1.15	2.16E-04	1.26	1.78E-03	1.26
1990	2.98E+04	1.17	1.35E-03	1.99	3.45E-04	1.82	5.47E-04	1.82
1991	2.31E+04	1.33	1.06E-03	1.99	7.74E-04	1.82	2.84E-04	1.82
1992	1.84E+04	1.20	4.35E-06	1.99	9.10E-05	1.82	1.59E-04	1.82
1993	2.21E+04	2.34	5.31E-04	1.99	2.02E-04	1.82	1.93E-04	1.82
1994	1.76E+04	1.82	2.32E-06	1.99	2.20E-04	1.82	2.42E-04	1.82
1995	1.17E+04	1.51	4.94E-04	1.99	1.62E-04	1.82	1.60E-04	1.82
1996	7.00E+03	1.58	1.09E-03	1.99	7.29E-05	1.82	1.73E-04	1.82
1997	7.81E+03	1.50	2.43E-04	1.99	6.33E-06	1.82	1.05E-05	1.82
1998	8.46E+03	1.44	2.20E-06	1.99	1.86E-05	1.82	4.95E-06	1.82
1999	1.62E+03	1.73	2.20E-06	1.99	6.95E-05	1.82	5.14E-06	1.82
2000	1.81E+03	1.60	1.92E-03	1.99	6.86E-05	1.82	1.09E-05	1.82
2001	1.83E+03	1.46	2.32E-03	1.99	1.47E-05	1.82	1.40E-05	1.82

Table C-11 L Reactor Area Annual Intakes and geometric standard deviations, (Bq per year)

	Tritium		Iodine-131		Plutonium		Uranium	
Year	50th Percentile	GSD	50th Percentile	GSD	50th Percentile	GSD	50th Percentile	GSD
1955	5.72E+03	2.71	9.81E+01	1.90	8.19E-01	1.82	9.94E-02	1.82
1956	4.07E+04	1.77	2.35E+03	1.68	3.15E-03	1.82	3.61E-02	1.82
1957	1.54E+05	1.64	5.25E+02	2.02	4.23E-03	1.82	6.08E-04	1.82
1958	2.11E+05	1.72	6.89E+01	2.94	2.06E-03	1.26	4.47E-04	1.26
1959	1.18E+05	2.09	2.84E+02	2.10	2.36E-03	1.26	1.91E-03	1.26
1960	1.30E+05	2.29	2.25E+01	2.93	6.95E-03	1.26	5.34E-03	1.26
1961	1.23E+05	2.44	1.51E+02	2.10	1.19E-03	1.26	2.76E-03	1.26
1962	1.66E+05	2.49	2.49E+01	3.35	1.01E-03	1.26	1.95E-03	1.26
1963	1.44E+05	2.80	1.54E+01	3.48	3.09E-04	1.26	6.99E-03	1.26
1964	3.14E+05	3.12	1.43E+01	3.07	5.15E-04	1.26	1.24E-02	1.26
1965	2.00E+05	2.98	7.72E+00	2.90	1.56E-03	1.26	1.13E-02	1.26
1966	1.63E+05	2.96	4.42E+00	2.20	9.50E-04	1.26	4.50E-03	1.26
1967	1.58E+05	3.03	1.28E+01	2.94	9.99E-04	1.26	4.18E-03	1.26
1968	1.33E+05	3.18	3.73E+00	2.52	6.21E-04	1.26	6.26E-03	1.26
1969	6.99E+04	1.29	4.96E+00	2.30	1.51E-01	1.26	1.58E-02	1.26
1970	8.06E+04	1.28	3.00E+00	1.71	2.83E-03	1.26	3.69E-03	1.26
1971	9.45E+04	1.27	1.81E+00	1.34	2.84E-03	1.26	1.19E-03	1.26
1972	9.96E+04	1.25	2.75E-01	1.96	1.96E-03	1.26	1.75E-03	1.26
1973	8.11E+04	1.27	1.64E-01	1.70	2.27E-03	1.26	1.15E-03	1.26
1974	9.95E+04	1.43	1.34E-01	1.18	7.24E-04	1.26	1.88E-03	1.26
1975	6.56E+04	1.35	8.96E-03	1.28	2.40E-04	1.26	1.03E-03	1.26
1976	4.58E+04	1.27	1.09E-02	1.19	1.58E-03	1.26	1.08E-03	1.26
1977	4.88E+04	1.28	4.64E-03	1.38	4.93E-04	1.26	3.45E-04	1.26
1978	4.50E+04	1.29	4.74E-03	1.29	6.91E-04	1.26	6.78E-04	1.26
1979	4.59E+04	1.28	8.06E-03	1.90	1.65E-04	1.26	5.39E-04	1.26
1980	4.30E+04	1.27	3.65E-03	2.35	3.81E-04	1.26	8.87E-04	1.26
1981	4.67E+04	1.27	5.02E-03	2.01	7.01E-04	1.26	1.37E-03	1.26
1982	5.25E+04	1.27	1.26E-02	2.14	5.11E-04	1.26	2.00E-03	1.26
1983	7.29E+04	1.31	8.15E-03	1.86	3.12E-04	1.26	9.80E-04	1.26
1984	8.36E+04	1.27	2.42E-02	1.68	1.65E-04	1.26	4.86E-04	1.26
1985	7.63E+04	1.30	7.79E-03	2.18	9.28E-05	1.26	5.59E-04	1.26
1986	5.07E+04	1.72	2.80E-03	1.95	2.16E-04	1.26	3.25E-04	1.26
1987	6.11E+04	1.64	6.65E-03	2.90	2.12E-04	1.26	1.85E-03	1.26
1988	4.57E+04	1.70	3.42E-03	3.16	1.03E-04	1.26	3.21E-04	1.26
1989	3.72E+04	1.98	2.52E-05	1.15	1.36E-04	1.26	1.10E-03	1.26
1990	3.20E+04	2.00	8.35E-04	1.99	2.24E-04	1.82	3.55E-04	1.82
1991	2.38E+04	2.20	6.55E-04	1.99	5.02E-04	1.82	1.84E-04	1.82
1992	1.90E+04	1.82	2.96E-06	1.99	5.93E-05	1.82	9.85E-05	1.82
1993	1.94E+04	3.54	3.44E-04	1.99	1.32E-04	1.82	1.20E-04	1.82
1994	1.45E+04	2.78	1.58E-06	1.99	1.46E-04	1.82	1.50E-04	1.82
1995	8.48E+03	2.33	3.16E-04	1.99	1.03E-04	1.82	9.87E-05	1.82
1996	5.43E+03	2.88	6.84E-04	1.99	4.74E-05	1.82	1.07E-04	1.82
1997	5.29E+03	2.41	1.65E-04	1.99	4.17E-06	1.82	7.00E-06	1.82
1998	5.48E+03	2.05	1.35E-06	1.99	1.23E-05	1.82	3.07E-06	1.82
1999	1.43E+03	3.27	1.35E-06	1.99	4.66E-05	1.82	3.21E-06	1.82
2000	1.54E+03	3.37	1.29E-03	1.99	4.57E-05	1.82	6.75E-06	1.82
2001	1.61E+03	2.49	1.55E-03	1.99	9.91E-06	1.82	8.62E-06	1.82

Table C-12 P Reactor Area Annual Intakes and geometric standard deviations, (Bq per year)

	Tritium		Iodine-131		Plutonium		Uranium	
Year	50th Percentile	GSD	50th Percentile	GSD	50th Percentile	GSD	50th Percentile	GSD
1955	1.73E+04	1.43	8.85E+01	1.91	7.04E-01	1.82	9.26E-02	1.82
1956	1.01E+05	1.31	2.19E+03	1.69	2.67E-03	1.82	3.04E-02	1.82
1957	2.23E+05	1.27	5.25E+02	2.05	3.60E-03	1.82	5.18E-04	1.82
1958	2.74E+05	1.26	5.73E+01	2.82	1.75E-03	1.26	3.99E-04	1.26
1959	2.49E+05	1.31	2.47E+02	2.13	2.01E-03	1.26	1.60E-03	1.26
1960	4.08E+05	1.32	2.35E+01	3.06	5.79E-03	1.26	4.45E-03	1.26
1961	4.11E+05	1.32	1.16E+02	1.99	1.03E-03	1.26	2.35E-03	1.26
1962	6.30E+05	1.32	2.35E+02	3.43	9.18E-04	1.26	1.66E-03	1.26
1963	4.36E+05	1.37	1.44E+01	3.48	2.66E-04	1.26	6.03E-03	1.26
1964	6.41E+05	1.37	1.80E+00	2.59	4.79E-04	1.26	1.08E-02	1.26
1965	5.07E+05	1.34	1.69E+00	2.05	1.50E-03	1.26	9.39E-03	1.26
1966	3.70E+05	1.34	2.16E+00	1.44	9.05E-04	1.26	3.79E-03	1.26
1967	3.40E+05	1.35	2.50E+00	2.32	9.49E-04	1.26	3.51E-03	1.26
1968	2.30E+05	1.37	5.55E+00	2.88	5.81E-04	1.26	5.85E-03	1.26
1969	2.55E+05	1.29	9.40E+00	2.82	1.49E-01	1.26	1.44E-02	1.26
1970	3.24E+05	1.30	2.50E+00	1.71	2.74E-03	1.26	3.17E-03	1.26
1971	3.17E+05	1.28	1.80E+00	1.72	2.75E-03	1.26	1.08E-03	1.26
1972	3.39E+05	1.27	4.00E-01	2.56	1.92E-03	1.26	1.55E-03	1.26
1973	4.46E+05	1.26	2.63E-01	2.47	2.23E-03	1.26	1.01E-03	1.26
1974	4.48E+05	1.32	1.11E-01	1.17	6.89E-04	1.26	1.57E-03	1.26
1975	4.32E+05	1.29	7.32E-03	1.22	2.35E-04	1.26	8.54E-04	1.26
1976	3.25E+05	1.26	9.03E-03	1.18	1.54E-03	1.26	9.03E-04	1.26
1977	2.42E+05	1.29	3.79E-03	1.33	4.81E-04	1.26	2.99E-04	1.26
1978	2.20E+05	1.29	3.87E-03	1.23	6.76E-04	1.26	5.83E-04	1.26
1979	2.69E+05	1.29	3.33E-02	3.05	1.53E-04	1.26	4.63E-04	1.26
1980	2.08E+05	1.28	2.51E-02	3.08	3.68E-04	1.26	7.60E-04	1.26
1981	1.81E+05	1.27	2.05E-02	2.96	6.75E-04	1.26	1.20E-03	1.26
1982	1.75E+05	1.27	6.74E-02	3.01	4.51E-04	1.26	1.69E-03	1.26
1983	2.16E+05	1.29	3.18E-02	2.89	2.94E-04	1.26	8.24E-04	1.26
1984	3.02E+05	1.30	7.85E-02	2.90	1.48E-04	1.26	4.19E-04	1.26
1985	2.64E+05	1.29	4.94E-02	2.97	8.15E-05	1.26	4.80E-04	1.26
1986	2.65E+05	1.28	6.56E-03	2.69	2.09E-04	1.26	2.79E-04	1.26
1987	3.29E+05	1.29	4.99E-02	3.08	2.04E-04	1.26	1.54E-03	1.26
1988	3.07E+05	1.29	2.73E-04	3.01	9.02E-05	1.26	2.71E-04	1.26
1989	3.91E+05	1.06	2.09E-05	1.15	1.17E-04	1.26	9.20E-04	1.26
1990	2.27E+05	1.07	6.94E-04	1.99	2.04E-04	1.82	3.24E-04	1.82
1991	5.99E+04	1.15	5.45E-04	1.99	4.59E-04	1.82	1.68E-04	1.82
1992	1.41E+04	1.21	2.92E-06	1.99	5.46E-05	1.82	8.28E-05	1.82
1993	1.89E+04	2.31	3.15E-04	1.99	1.21E-04	1.82	1.01E-04	1.82
1994	4.66E+04	1.56	1.56E-06	1.99	1.39E-04	1.82	1.27E-04	1.82
1995	3.41E+04	1.46	2.82E-04	1.99	9.10E-05	1.82	8.23E-05	1.82
1996	2.67E+04	1.47	5.87E-04	1.99	4.35E-05	1.82	8.88E-05	1.82
1997	1.18E+04	1.49	1.63E-04	1.99	3.91E-06	1.82	6.63E-06	1.82
1998	1.45E+04	1.44	1.13E-06	1.99	1.16E-05	1.82	2.57E-06	1.82
1999	3.24E+02	3.18	1.13E-06	1.99	4.48E-05	1.82	2.73E-06	1.82
2000	2.90E+02	3.08	1.25E-03	1.99	4.37E-05	1.82	5.70E-06	1.82
2001	2.91E+02	2.57	1.48E-03	1.99	9.68E-06	1.82	7.18E-06	1.82

Table C-13 K Reactor Area Annual Intakes and geometric standard deviations, (Bq per year)

	Tritium		Iodine-131		Plutonium		Uranium	
Year	50th Percentile	GSD	50th Percentile	GSD	50th Percentile	GSD	50th Percentile	GSD
1955	1.24E+04	1.46	1.09E+02	1.88	9.56E-01	1.82	1.06E-01	1.82
1956	6.46E+04	1.32	2.51E+03	1.68	3.73E-03	1.82	4.31E-02	1.82
1957	2.54E+05	1.27	5.26E+02	2.03	5.00E-03	1.82	7.18E-04	1.82
1958	2.67E+05	1.25	6.93E+01	2.95	2.45E-03	1.26	5.00E-04	1.26
1959	3.03E+05	1.31	3.26E+02	2.07	2.79E-03	1.26	2.30E-03	1.26
1960	4.56E+05	1.30	3.11E+01	3.01	8.37E-03	1.26	6.43E-03	1.26
1961	3.11E+05	1.31	1.73E+02	2.03	1.36E-03	1.26	3.25E-03	1.26
1962	5.37E+05	1.32	1.84E+01	3.29	1.12E-03	1.26	2.30E-03	1.26
1963	1.57E+05	3.30	7.31E+00	3.41	3.59E-04	1.26	8.13E-03	1.26
1964	7.72E+05	1.37	2.32E+00	2.51	5.51E-04	1.26	1.43E-02	1.26
1965	8.84E+05	1.32	3.41E+00	2.40	1.61E-03	1.26	1.36E-02	1.26
1966	6.70E+05	1.32	3.55E+00	1.68	9.83E-04	1.26	5.37E-03	1.26
1967	8.13E+05	1.34	1.43E+01	2.93	1.04E-03	1.26	5.00E-03	1.26
1968	7.13E+05	1.32	1.65E+01	3.10	6.58E-04	1.26	6.63E-03	1.26
1969	6.13E+05	1.30	2.67E+01	3.00	1.49E-01	1.26	1.73E-02	1.26
1970	7.64E+05	1.30	8.19E+00	2.61	2.87E-03	1.26	4.30E-03	1.26
1971	8.22E+05	1.29	2.21E+00	1.37	2.86E-03	1.26	1.30E-03	1.26
1972	7.37E+05	1.27	5.26E-01	2.48	1.96E-03	1.26	1.97E-03	1.26
1973	6.52E+05	1.28	4.09E-01	2.52	2.26E-03	1.26	1.31E-03	1.26
1974	5.39E+05	1.31	1.63E-01	1.21	7.51E-04	1.26	2.27E-03	1.26
1975	3.92E+05	1.31	1.51E-02	1.92	2.40E-04	1.26	1.24E-03	1.26
1976	4.12E+05	1.25	1.32E-02	1.21	1.59E-03	1.26	1.30E-03	1.26
1977	2.72E+05	1.28	5.98E-03	1.53	4.94E-04	1.26	4.00E-04	1.26
1978	2.84E+05	1.29	7.81E-03	1.89	6.89E-04	1.26	7.90E-04	1.26
1979	2.87E+05	1.29	8.71E-03	1.69	1.75E-04	1.26	6.30E-04	1.26
1980	3.37E+05	1.27	4.37E-03	2.35	3.88E-04	1.26	1.04E-03	1.26
1981	3.76E+05	1.27	5.41E-03	1.85	7.15E-04	1.26	1.55E-03	1.26
1982	3.72E+05	1.28	1.69E-02	2.25	5.79E-04	1.26	2.38E-03	1.26
1983	5.31E+05	1.31	8.61E-03	1.63	3.28E-04	1.26	1.17E-03	1.26
1984	3.82E+05	1.28	2.73E-02	1.54	1.84E-04	1.26	5.65E-04	1.26
1985	3.60E+05	1.28	9.56E-03	2.20	1.06E-04	1.26	6.53E-04	1.26
1986	2.65E+05	1.29	9.86E-03	2.70	2.20E-04	1.26	3.81E-04	1.26
1987	2.37E+05	1.29	5.24E-03	2.79	2.17E-04	1.26	2.22E-03	1.26
1988	2.45E+05	1.29	3.25E-04	2.97	1.18E-04	1.26	3.81E-04	1.26
1989	1.17E+05	1.09	3.04E-05	1.15	1.58E-04	1.26	1.32E-03	1.26
1990	1.63E+05	1.01	1.01E-03	1.99	2.44E-04	1.82	3.87E-04	1.82
1991	2.08E+05	0.98	7.91E-04	1.99	5.48E-04	1.82	2.01E-04	1.82
1992	1.72E+05	1.07	2.92E-06	1.99	6.41E-05	1.82	1.18E-04	1.82
1993	1.55E+05	5.03	3.76E-04	1.99	1.43E-04	1.82	1.43E-04	1.82
1994	5.83E+04	4.68	1.56E-06	1.99	1.52E-04	1.82	1.79E-04	1.82
1995	1.65E+04	3.88	3.54E-04	1.99	1.17E-04	1.82	1.19E-04	1.82
1996	1.31E+04	4.27	8.00E-04	1.99	5.14E-05	1.82	1.29E-04	1.82
1997	1.17E+04	3.93	1.63E-04	1.99	4.40E-06	1.82	7.29E-06	1.82
1998	1.00E+04	3.58	1.64E-06	1.99	1.29E-05	1.82	3.67E-06	1.82
1999	7.01E+03	4.32	1.64E-06	1.99	4.75E-05	1.82	3.78E-06	1.82
2000	5.02E+03	4.40	1.31E-03	1.99	4.72E-05	1.82	8.01E-06	1.82
2001	3.32E+03	4.19	1.59E-03	1.99	9.93E-06	1.82	1.04E-05	1.82

Table C-14 R Reactor Area Annual Intakes and geometric standard deviations, (Bq per year)

	Tritium		Iodine-131		Plutonium		Uranium	
Year	50th Percentile	GSD	50th Percentile	GSD	50th Percentile	GSD	50th Percentile	GSD
1955	4.12E+03	2.09	1.31E+02	1.86	9.76E-01	1.82	1.46E-01	1.82
1956	4.14E+04	1.75	3.44E+03	1.68	3.61E-03	1.82	4.05E-02	1.82
1957	2.05E+05	1.44	8.55E+02	1.98	4.89E-03	1.82	7.03E-04	1.82
1958	2.89E+05	1.57	7.54E+01	2.58	2.36E-03	1.26	5.92E-04	1.26
1959	1.49E+05	1.94	3.26E+02	2.05	2.75E-03	1.26	2.09E-03	1.26
1960	1.47E+05	2.27	3.55E+01	3.13	7.55E-03	1.26	5.82E-03	1.26
1961	1.20E+05	1.84	1.65E+02	2.12	1.47E-03	1.26	3.20E-03	1.26
1962	1.73E+05	2.40	2.58E+01	3.34	1.39E-03	1.26	2.26E-03	1.26
1963	1.63E+05	2.37	1.01E+02	3.51	3.71E-04	1.26	8.41E-03	1.26
1964	2.16E+05	2.46	1.72E+00	2.37	7.54E-04	1.26	1.54E-02	1.26
1965	7.96E+04	1.40	1.66E+00	1.60	2.47E-03	1.26	1.23E-02	1.26
1966	6.91E+04	1.39	2.63E+00	1.27	1.48E-03	1.26	5.03E-03	1.26
1967	7.09E+04	1.38	2.24E+00	1.90	1.54E-03	1.26	4.65E-03	1.26
1968	8.24E+04	1.35	2.39E+00	1.87	9.25E-04	1.26	9.31E-03	1.26
1969	5.65E+04	1.26	3.69E+00	1.83	2.55E-01	1.26	2.19E-02	1.26
1970	6.08E+04	1.25	2.84E+00	1.42	4.58E-03	1.26	4.41E-03	1.26
1971	7.88E+04	1.24	1.92E+00	1.27	4.61E-03	1.26	1.65E-03	1.26
1972	9.58E+04	1.23	2.57E-01	1.71	3.25E-03	1.26	2.26E-03	1.26
1973	6.93E+04	1.24	1.57E-01	1.45	3.81E-03	1.26	1.47E-03	1.26
1974	1.07E+05	1.51	1.45E-01	1.17	1.12E-03	1.26	2.04E-03	1.26
1975	6.30E+04	1.38	9.52E-03	1.19	4.00E-04	1.26	1.11E-03	1.26
1976	3.50E+04	1.26	1.17E-02	1.18	2.59E-03	1.26	1.18E-03	1.26
1977	4.39E+04	1.26	4.88E-03	1.29	8.12E-04	1.26	4.21E-04	1.26
1978	4.04E+04	1.28	4.99E-03	1.20	1.15E-03	1.26	8.10E-04	1.26
1979	3.95E+04	1.25	8.11E-03	1.75	2.43E-04	1.26	6.40E-04	1.26
1980	3.45E+04	1.25	3.36E-03	2.18	6.12E-04	1.26	1.05E-03	1.26
1981	4.37E+04	1.25	4.86E-03	1.85	1.12E-03	1.26	1.74E-03	1.26
1982	4.92E+04	1.24	1.19E-02	1.96	6.56E-04	1.26	2.26E-03	1.26
1983	7.19E+04	1.25	8.12E-03	1.72	4.72E-04	1.26	1.09E-03	1.26
1984	9.04E+04	1.24	2.48E-02	1.57	2.21E-04	1.26	5.83E-04	1.26
1985	7.36E+04	1.25	7.28E-03	2.01	1.17E-04	1.26	6.63E-04	1.26
1986	4.97E+04	1.24	2.45E-03	1.62	3.47E-04	1.26	3.85E-04	1.26
1987	7.25E+04	1.32	3.90E-03	2.72	3.38E-04	1.26	2.01E-03	1.26
1988	4.76E+04	1.25	1.54E-04	2.78	1.29E-04	1.26	3.63E-04	1.26
1989	3.82E+04	1.17	2.72E-05	1.15	1.64E-04	1.26	1.21E-03	1.26
1990	3.27E+04	1.18	9.02E-04	1.99	3.14E-04	1.82	4.98E-04	1.82
1991	2.19E+04	1.40	7.08E-04	1.99	7.05E-04	1.82	2.58E-04	1.82
1992	1.85E+04	1.21	5.01E-06	1.99	8.49E-05	1.82	1.10E-04	1.82
1993	2.10E+04	2.03	4.83E-04	1.99	1.87E-04	1.82	1.34E-04	1.82
1994	1.86E+04	1.66	2.67E-06	1.99	2.25E-04	1.82	1.70E-04	1.82
1995	1.15E+04	1.50	4.17E-04	1.99	1.32E-04	1.82	1.07E-04	1.82
1996	6.34E+03	1.57	8.10E-04	1.99	6.72E-05	1.82	1.15E-04	1.82
1997	6.79E+03	1.51	2.79E-04	1.99	6.25E-06	1.82	1.08E-05	1.82
1998	7.56E+03	1.47	1.46E-06	1.99	1.86E-05	1.82	3.38E-06	1.82
1999	3.99E+02	2.51	1.46E-06	1.99	7.44E-05	1.82	3.72E-06	1.82
2000	3.39E+02	2.46	2.08E-03	1.99	7.16E-05	1.82	7.66E-06	1.82
2001	3.67E+02	2.07	2.45E-03	1.99	1.63E-05	1.82	9.37E-06	1.82

Table C-15 Rail Yard Annual Intakes and geometric standard deviations, (Bq per year)

	Tritium		Iodine-131		Plutonium		Uranium	
Year	50th Percentile	GSD	50th Percentile	GSD	50th Percentile	GSD	50th Percentile	GSD
1955	2.80E+03	1.53	6.67E+01	1.82	5.47E-01	1.82	7.13E-02	1.82
1956	2.55E+04	1.32	1.68E+03	1.67	2.08E-03	1.82	2.37E-02	1.82
1957	1.01E+05	1.27	3.74E+02	1.92	2.80E-03	1.82	4.02E-04	1.82
1958	1.35E+05	1.26	2.84E+01	2.37	1.36E-03	1.26	3.09E-04	1.26
1959	6.79E+04	1.34	1.72E+02	1.92	1.56E-03	1.26	1.24E-03	1.26
1960	7.16E+04	1.39	9.78E+00	2.60	4.51E-03	1.26	3.47E-03	1.26
1961	6.79E+04	1.36	8.49E+01	1.88	8.02E-04	1.26	1.83E-03	1.26
1962	9.11E+04	1.42	2.74E+01	3.38	7.09E-04	1.26	1.29E-03	1.26
1963	7.50E+04	1.51	6.58E+00	3.45	2.07E-04	1.26	4.68E-03	1.26
1964	1.09E+05	1.55	1.05E+00	2.38	3.69E-04	1.26	8.39E-03	1.26
1965	6.82E+04	1.46	1.12E+00	1.83	1.15E-03	1.26	7.31E-03	1.26
1966	5.45E+04	1.46	1.62E+00	1.35	6.95E-04	1.26	2.94E-03	1.26
1967	5.50E+04	1.45	1.64E+00	2.15	7.29E-04	1.26	2.73E-03	1.26
1968	5.36E+04	1.43	1.83E+00	2.23	4.47E-04	1.26	4.50E-03	1.26
1969	4.23E+04	1.28	2.89E+00	2.19	1.14E-01	1.26	1.11E-02	1.26
1970	4.90E+04	1.28	1.77E+00	1.51	2.10E-03	1.26	2.46E-03	1.26
1971	5.70E+04	1.26	1.18E+00	1.35	2.11E-03	1.26	8.34E-04	1.26
1972	6.41E+04	1.25	1.74E-01	1.93	1.47E-03	1.26	1.20E-03	1.26
1973	5.74E+04	1.26	1.07E-01	1.72	1.71E-03	1.26	7.85E-04	1.26
1974	7.39E+04	1.44	8.65E-02	1.17	5.29E-04	1.26	1.22E-03	1.26
1975	5.33E+04	1.34	5.68E-03	1.21	1.80E-04	1.26	6.65E-04	1.26
1976	3.40E+04	1.27	7.02E-03	1.18	1.18E-03	1.26	7.03E-04	1.26
1977	3.49E+04	1.28	2.93E-03	1.32	3.69E-04	1.26	2.32E-04	1.26
1978	3.17E+04	1.29	3.00E-03	1.22	5.18E-04	1.26	4.53E-04	1.26
1979	3.45E+04	1.28	6.99E-03	2.30	1.18E-04	1.26	3.59E-04	1.26
1980	2.86E+04	1.27	3.81E-03	2.68	2.82E-04	1.26	5.90E-04	1.26
1981	3.03E+04	1.27	4.27E-03	2.32	5.18E-04	1.26	9.31E-04	1.26
1982	3.33E+04	1.26	1.18E-02	2.48	3.49E-04	1.26	1.31E-03	1.26
1983	4.58E+04	1.27	6.91E-03	2.20	2.26E-04	1.26	6.41E-04	1.26
1984	5.88E+04	1.27	1.96E-02	2.05	1.15E-04	1.26	3.25E-04	1.26
1985	5.04E+04	1.27	7.85E-03	2.52	6.31E-05	1.26	3.72E-04	1.26
1986	3.76E+04	1.27	1.89E-03	1.99	1.60E-04	1.26	2.17E-04	1.26
1987	4.99E+04	1.32	6.10E-03	2.96	1.57E-04	1.26	1.20E-03	1.26
1988	3.94E+04	1.28	1.53E-04	2.94	6.99E-05	1.26	2.11E-04	1.26
1989	3.99E+04	1.12	1.63E-05	1.15	9.10E-05	1.26	7.16E-04	1.26
1990	2.80E+04	1.13	5.41E-04	1.99	1.58E-04	1.82	2.50E-04	1.82
1991	1.39E+04	1.32	4.24E-04	1.99	3.54E-04	1.82	1.30E-04	1.82
1992	9.53E+03	1.21	2.24E-06	1.99	4.21E-05	1.82	6.44E-05	1.82
1993	1.07E+04	2.44	2.43E-04	1.99	9.32E-05	1.82	7.83E-05	1.82
1994	1.08E+04	1.80	1.19E-06	1.99	1.07E-04	1.82	9.85E-05	1.82
1995	6.81E+03	1.55	2.18E-04	1.99	7.04E-05	1.82	6.40E-05	1.82
1996	4.23E+03	1.63	4.55E-04	1.99	3.35E-05	1.82	6.91E-05	1.82
1997	3.53E+03	1.60	1.25E-04	1.99	3.01E-06	1.82	5.10E-06	1.82
1998	3.98E+03	1.53	8.77E-07	1.99	8.92E-06	1.82	2.00E-06	1.82
1999	2.27E+02	3.06	8.77E-07	1.99	3.44E-05	1.82	2.12E-06	1.82
2000	2.01E+02	2.96	9.56E-04	1.99	3.35E-05	1.82	4.43E-06	1.82
2001	2.04E+02	2.47	1.14E-03	1.99	7.42E-06	1.82	5.59E-06	1.82

Table C-16 Average Site-Wide [(a)] Annual Intakes and Geometric Standard Deviations, (Bq per year)

	Tritium		Iodine-131		Plutonium		Uranium	
Year	50th Percentile	GSD	50th Percentile	GSD	50th Percentile	GSD	50th Percentile	GSD
1955	2.86E+04	1.33E+00	4.31E+02	1.80E+00	3.64E+00	1.82E+00	4.53E-01	1.82E+00
1956	1.61E+05	1.27E+00	1.06E+04	1.66E+00	1.40E-02	1.82E+00	1.60E-01	1.82E+00
1957	6.29E+05	1.24E+00	2.26E+03	1.89E+00	1.88E-02	1.82E+00	2.70E-03	1.82E+00
1958	8.42E+05	1.23E+00	1.51E+02	2.16E+00	9.15E-03	1.26E+00	2.01E-03	1.26E+00
1959	4.15E+05	1.26E+00	1.12E+03	1.87E+00	1.05E-02	1.26E+00	8.43E-03	1.26E+00
1960	3.73E+05	1.31E+00	5.04E+01	2.24E+00	3.06E-02	1.26E+00	2.35E-02	1.26E+00
1961	3.40E+05	1.28E+00	5.61E+02	1.83E+00	5.30E-03	1.26E+00	1.22E-02	1.26E+00
1962	4.18E+05	1.35E+00	3.20E+01	3.13E+00	4.58E-03	1.26E+00	8.65E-03	1.26E+00
1963	3.82E+05	1.43E+00	1.48E+01	3.32E+00	1.37E-03	1.26E+00	3.11E-02	1.26E+00
1964	5.54E+05	1.44E+00	4.76E+00	1.93E+00	2.35E-03	1.26E+00	5.55E-02	1.26E+00
1965	3.27E+05	1.38E+00	6.48E+00	1.52E+00	7.19E-03	1.26E+00	4.96E-02	1.26E+00
1966	3.01E+05	1.35E+00	1.06E+01	1.25E+00	4.36E-03	1.26E+00	1.99E-02	1.26E+00
1967	3.04E+05	1.35E+00	8.67E+00	1.82E+00	4.58E-03	1.26E+00	1.85E-02	1.26E+00
1968	3.26E+05	1.34E+00	9.19E+00	1.76E+00	2.83E-03	1.26E+00	2.85E-02	1.26E+00
1969	2.34E+05	1.25E+00	1.41E+01	1.72E+00	6.99E-01	1.26E+00	7.14E-02	1.26E+00
1970	2.48E+05	1.24E+00	1.15E+01	1.41E+00	1.31E-02	1.26E+00	1.64E-02	1.26E+00
1971	2.95E+05	1.24E+00	7.75E+00	1.26E+00	1.31E-02	1.26E+00	5.38E-03	1.26E+00
1972	3.40E+05	1.23E+00	1.02E+00	1.65E+00	9.08E-03	1.26E+00	7.84E-03	1.26E+00
1973	2.64E+05	1.23E+00	6.19E-01	1.38E+00	1.05E-02	1.26E+00	5.16E-03	1.26E+00
1974	3.52E+05	1.47E+00	5.87E-01	1.17E+00	3.32E-03	1.26E+00	8.30E-03	1.26E+00
1975	2.15E+05	1.35E+00	3.82E-02	1.19E+00	1.11E-03	1.26E+00	4.52E-03	1.26E+00
1976	1.44E+05	1.24E+00	4.77E-02	1.18E+00	7.29E-03	1.26E+00	4.77E-03	1.26E+00
1977	1.65E+05	1.25E+00	1.97E-02	1.29E+00	2.28E-03	1.26E+00	1.54E-03	1.26E+00
1978	1.53E+05	1.27E+00	2.03E-02	1.20E+00	3.20E-03	1.26E+00	3.02E-03	1.26E+00
1979	1.45E+05	1.26E+00	2.90E-02	1.48E+00	7.50E-04	1.26E+00	2.40E-03	1.26E+00
1980	1.38E+05	1.24E+00	1.05E-02	1.82E+00	1.76E-03	1.26E+00	3.94E-03	1.26E+00
1981	1.61E+05	1.24E+00	1.74E-02	1.62E+00	3.23E-03	1.26E+00	6.12E-03	1.26E+00
1982	1.78E+05	1.24E+00	4.02E-02	1.66E+00	2.29E-03	1.26E+00	8.86E-03	1.26E+00
1983	2.49E+05	1.25E+00	2.94E-02	1.49E+00	1.43E-03	1.26E+00	4.33E-03	1.26E+00
1984	3.07E+05	1.25E+00	9.25E-02	1.36E+00	7.44E-04	1.26E+00	2.16E-03	1.26E+00
1985	2.74E+05	1.25E+00	2.35E-02	1.67E+00	4.15E-04	1.26E+00	2.48E-03	1.26E+00
1986	1.74E+05	1.24E+00	9.31E-03	1.49E+00	9.97E-04	1.26E+00	1.45E-03	1.26E+00
1987	2.19E+05	1.31E+00	9.22E-03	2.42E+00	9.77E-04	1.26E+00	8.14E-03	1.26E+00
1988	1.44E+05	1.25E+00	4.91E-04	2.67E+00	4.61E-04	1.26E+00	1.42E-03	1.26E+00
1989	1.06E+05	1.18E+00	1.11E-04	1.15E+00	6.05E-04	1.26E+00	4.85E-03	1.26E+00
1990	9.48E+04	1.17E+00	3.68E-03	1.99E+00	1.01E-03	1.82E+00	1.61E-03	1.82E+00
1991	6.96E+04	1.35E+00	2.89E-03	1.99E+00	2.27E-03	1.82E+00	8.34E-04	1.82E+00
1992	5.85E+04	1.20E+00	1.37E-05	1.99E+00	2.69E-04	1.82E+00	4.35E-04	1.82E+00
1993	6.50E+04	2.46E+00	1.56E-03	1.99E+00	5.97E-04	1.82E+00	5.30E-04	1.82E+00
1994	5.36E+04	1.87E+00	7.34E-06	1.99E+00	6.70E-04	1.82E+00	6.64E-04	1.82E+00
1995	3.27E+04	1.57E+00	1.42E-03	1.99E+00	4.62E-04	1.82E+00	4.35E-04	1.82E+00
1996	1.84E+04	1.69E+00	3.04E-03	1.99E+00	2.15E-04	1.82E+00	4.70E-04	1.82E+00
1997	1.98E+04	1.60E+00	7.67E-04	1.99E+00	1.90E-05	1.82E+00	3.21E-05	1.82E+00
1998	2.18E+04	1.52E+00	5.97E-06	1.99E+00	5.63E-05	1.82E+00	1.35E-05	1.82E+00
1999	2.01E+03	2.54E+00	5.97E-06	1.99E+00	2.14E-04	1.82E+00	1.42E-05	1.82E+00
2000	1.91E+03	2.36E+00	5.94E-03	1.99E+00	2.10E-04	1.82E+00	2.99E-05	1.82E+00
2001	1.92E+03	1.98E+00	7.12E-03	1.99E+00	4.58E-05	1.82E+00	3.80E-05	1.82E+00

(a) Values are for plant perimeter.

Table C-17 Maximum Site-Wide Annual Intakes and geometric standard deviations, (Bq per year)

	Tritium		Iodine-131		Plutonium		Uranium	
Year	50th Percentile	GSD	50th Percentile	GSD	50th Percentile	GSD	50th Percentile	GSD
1955	2.12E+05	1.18	2.06E+03	1.79	2.17E+00	1.82	2.46E+01	1.82
1956	1.02E+06	1.26	5.03E+04	1.66	1.22E+00	1.82	1.02E-01	1.82
1957	2.88E+06	1.24	1.50E+04	1.87	1.94E-02	1.82	1.35E-01	1.82
1958	3.93E+06	1.20	9.41E+02	1.98	1.01E-02	1.26	6.74E-02	1.26
1959	2.12E+06	1.22	7.58E+03	1.84	6.73E-02	1.26	7.48E-02	1.26
1960	1.68E+06	1.22	3.57E+02	2.00	1.89E-01	1.26	2.47E-01	1.26
1961	1.72E+06	1.20	4.50E+03	1.80	8.75E-02	1.26	3.26E-02	1.26
1962	2.01E+06	1.24	2.35E+02	3.43	6.20E-02	1.26	2.06E-02	1.26
1963	1.97E+06	1.22	1.01E+02	3.51	2.05E-01	1.26	9.06E-03	1.26
1964	2.54E+06	1.28	2.69E+01	2.09	3.39E-01	1.26	1.10E-02	1.26
1965	1.00E+06	1.23	4.62E+01	1.18	3.99E-01	1.26	4.19E-02	1.26
1966	1.01E+06	1.23	8.21E+01	1.12	1.53E-01	1.26	2.45E-02	1.26
1967	9.22E+05	1.24	5.16E+01	1.34	1.43E-01	1.26	2.52E-02	1.26
1968	1.27E+06	1.22	5.91E+01	1.43	1.41E-01	1.26	1.40E-02	1.26
1969	8.03E+05	1.21	9.07E+01	1.42	3.07E-01	1.26	4.82E+00	1.26
1970	8.58E+05	1.21	8.60E+01	1.17	1.10E-01	1.26	8.10E-02	1.26
1971	1.10E+06	1.21	6.09E+01	1.16	2.31E-02	1.26	8.23E-02	1.26
1972	1.47E+06	1.20	6.88E+00	1.21	4.24E-02	1.26	5.97E-02	1.26
1973	1.00E+06	1.19	4.63E+00	1.17	2.91E-02	1.26	7.10E-02	1.26
1974	1.68E+06	1.53	4.72E+00	1.15	6.73E-02	1.26	1.84E-02	1.26
1975	8.69E+05	1.39	2.99E-01	1.16	3.67E-02	1.26	7.37E-03	1.26
1976	4.69E+05	1.20	3.85E-01	1.16	3.81E-02	1.26	4.66E-02	1.26
1977	5.77E+05	1.21	1.50E-01	1.17	9.85E-03	1.26	1.48E-02	1.26
1978	5.58E+05	1.24	1.62E-01	1.16	2.02E-02	1.26	2.11E-02	1.26
1979	5.06E+05	1.20	2.08E-01	1.17	1.63E-02	1.26	3.61E-03	1.26
1980	4.50E+05	1.19	6.24E-02	1.77	2.71E-02	1.26	1.07E-02	1.26
1981	7.13E+05	1.21	1.18E-01	1.19	3.49E-02	1.26	1.94E-02	1.26
1982	7.41E+05	1.20	2.65E-01	1.31	6.68E-02	1.26	1.27E-02	1.26
1983	1.16E+06	1.22	2.11E-01	1.17	3.35E-02	1.26	7.37E-03	1.26
1984	1.41E+06	1.21	7.01E-01	1.16	1.43E-02	1.26	3.63E-03	1.26
1985	1.05E+06	1.21	1.52E-01	1.50	1.69E-02	1.26	2.43E-03	1.26
1986	8.77E+05	1.19	6.60E-02	1.17	9.88E-03	1.26	6.06E-03	1.26
1987	1.28E+06	1.30	4.99E-02	3.08	6.55E-02	1.26	5.80E-03	1.26
1988	7.17E+05	1.21	3.42E-03	3.16	1.07E-02	1.26	2.72E-03	1.26
1989	4.38E+05	1.21	9.05E-04	1.15	3.84E-02	1.26	3.96E-03	1.26
1990	4.33E+05	1.20	3.00E-02	1.99	6.58E-03	1.82	4.15E-03	1.82
1991	3.28E+05	1.46	2.36E-02	1.99	3.42E-03	1.82	9.31E-03	1.82
1992	3.06E+05	1.21	9.47E-05	1.99	3.37E-03	1.82	1.18E-03	1.82
1993	3.54E+05	3.05	6.40E-03	1.99	4.09E-03	1.82	2.56E-03	1.82
1994	3.16E+05	1.68	5.06E-05	1.99	5.00E-03	1.82	3.66E-03	1.82
1995	1.95E+05	1.41	7.31E-03	1.99	3.52E-03	1.82	2.58E-03	1.82
1996	1.01E+05	1.42	2.08E-02	1.99	3.84E-03	1.82	9.11E-04	1.82
1997	1.21E+05	1.41	5.27E-03	1.99	1.74E-04	1.82	9.60E-05	1.82
1998	1.35E+05	1.41	4.88E-05	1.99	1.06E-04	1.82	2.92E-04	1.82
1999	7.01E+03	4.32	4.88E-05	1.99	1.02E-04	1.82	1.29E-03	1.82
2000	5.80E+03	4.03	3.66E-02	1.99	2.23E-04	1.82	1.20E-03	1.82
2001	5.78E+03	2.99	4.14E-02	1.99	3.08E-04	1.82	2.98E-04	1.82

C.1.2 Resuspension of Radioactive Materials from Soil

Soil analyses indicated that elevated levels of certain radionuclides. including plutonium, were detected at several locations near the release points in both F and H Areas. It was concluded that the measured soil concentrations in F and H Areas were due to deposition during the years of the greatest releases. Annual intakes based on ventilation rate of 2,400 cubic meters per year resulting from resuspension of radionuclides in soil are presented in Table C-18.

Table C-18 Annual Intakes for F and H Areas (Bq per year)

	^{90}Sr		^{137}Cs		^{238}Pu		^{239}Pu	
Location	**50th**	**GSD**	**50th**	**GSD**	**50th**	**GSD**	**50th**	**GSD**
F Area	2.46E-04	1.81	1.19E-02	1.06	4.59E-03	1.04	6.04E-03	1.03
H Area	3.29E-04	1.63	1.50E-02	1.04	2.42E-04	1.09	6.30E-04	1.05

For purposes of reconstructing potential missed or unmonitored dose, the annual intakes provided in Table C-18 should be used only for individuals whose employment records indicate that they spent considerable time in either F or H Areas. The ^{239}Pu annual intakes should be assumed to have persisted since 1955 and the ^{238}Pu annual intakes since 1969. Annual intakes of ^{90}Sr and ^{137}Cs should be conservatively assumed to have persisted essentially unchanged since 1955.

C.2 External Dose

C.2.1 Ambient Radiation

The ambient radiation levels in Table C-19 are presented in mrem/year for various locations at the SRP. These levels are for a full work year of 2,000 hours which represent a fraction (i.e., 0.228) of the full annual doses (i.e., 8766 hours) that were derived from the SRS Annual Environmental Reports. The values in Table C-19 include the contributions from natural background radiation and nuclear weapons fallout. Therefore, use of these data to account for potential missed or unmonitored dose should in all cases represent an overestimation of the worker's actual dose.

Table C-19 Ambient radiation levels for selected SRS locations (mrem/year)

	A Area		C Area		D-Area		E Area		F Area		H Area		S Area		Z Area	
Year	50th	GSD	50th	GSD	50th	GSD	50th	GSD	50th	GSD	50th	GSD	50th	GSD	50th	GSD
1954	73		62		58						62					
1955	78		69		75				75		74					
1956	75		74		69				77		90					
1957	96		86		80				90		107					
1958	102		103		98				98		120					
1959	59		56		54				88		77					
1960	61		41		40				182		81					
1961	44		41		38				49		57					
1962	64		47		50				49		61					
1963	68		50		57				60		81					
1964	35		48		30				58		75					
1965	33		48		28				48		63					
1966	24				22				49		105					
1967	23				20				38		89					
1968	18				18				40		48					
1969	20				23				51		67					
1970	19				13				56		81					
1971	19				14				68		47					
1972	22		17		12				76		47					
1973	26	2.51	19	2.77	8	4.34			79	4.34	55	4.58				
1974	22	1.19	18	1.23	23	1.77			168	1.10	128	1.10				
1975	58	1.22	21	1.29	18	1.23			168	1.13	133	1.16				
1976	128	1.17	23	1.26	18	1.24			140	1.17	142	1.12				
1977	192	1.15	21	1.20	20	1.21			175	1.16	183	1.16				
1978	43	1.19	21	1.20	21	1.20			100	1.17	83	1.18				
1979	53	1.16	31	1.20	15	1.27			129	1.13	93	1.18				
1980	34	1.18	18	1.23	17	1.25			80	1.08	56	1.19				
1981	55	1.19	23	1.26	22	1.19			71	1.15	180	1.07				
1982	42	1.10	28	1.22	20	1.21	27	1.23	65	1.19	142	1.16				
1983	39	1.21	20	1.21	18	1.23	31	1.20	63	1.20	133	1.16				
1984	27	1.23	23	1.19	21	1.20	39	1.21	70	1.18	125	1.17	22	1.19		
1985	33	1.19	18	1.24	20	1.21	41	1.20	48	1.18	467	1.15	48	1.18	18	1.23
1986	37	1.23	21	1.20	22	1.19	29	1.21	60	1.18	117	1.16	46	1.18	18	1.23
1987	26	1.24	18	1.24	21	1.20	29	1.21	47	1.18	92	1.16	30	1.21	18	1.24
1988	27	1.44	20	1.40	18	1.43	27	1.44	43	1.41	167	1.41	27	1.44	19	1.41
1989	33	1.41	18	1.45	18	1.43	27	1.44	37	1.43	79	1.42	27	1.44	18	1.45
1990	21	1.42	12	1.41	16	1.42	182	1.43	34	1.43	55	1.42	23	1.43	13	1.42
1991	20	1.42	11	1.42	16	1.42	110	1.42	30	1.43	35	1.43	23	1.42	12	1.42
1992	18	1.48	10	1.11	13	1.11	76	1.11	24	1.11	29	1.11	20	1.11	13	1.39
1993	24	1.11	16	1.11	18	1.11	79	1.11	29	1.11	33	1.11	27	1.11	18	1.11
1994	24	1.11	17	1.11	18	1.11	78	1.11	31	1.11	36	1.11	28	1.11	19	1.11
1995	25	1.11	18	1.11	19	1.11	56	1.11	32	1.11	34	1.11	29	1.11	20	1.11
1996	26	1.11	19	1.11	19	1.11	55	1.11	31	1.11	34	1.11	28	1.11	19	1.11
1997	26	1.11					42	1.11	31	1.11	34	1.11	27	1.11	20	1.11
1998	27	1.11					35	1.11	33	1.11	33	1.11	29	1.11	22	1.11
1999					18	1.11										
2000					19	1.11										
2001					17	1.11										

Table C-19 Continued

	Forest Service[(a)]		Central Shops		L-Area		P Area		K Area		R Area		Rail Yard		Site Perimeter	
Year	50th	GSD	50th	GSD	50th	GSD	50th	GSD	50th	GSD	50th	GSD	50th	GSD	50th	GSD
1954							67				77					
1955					84		83		80		75					
1956	69				78		70		75		73				69	
1957	83				95		92		91		93				83	
1958	90				110		104		106		103				90	
1959	41				61		56		55		64				41	
1960	30				48		42		47		47				30	
1961	26				38		36		43		33				26	
1962	37				50		49		49		49				37	
1963	36				59		58		57		58				36	
1964	32				52		43		45		44				32	
1965	28														28	
1966	21														21	
1967	19														19	
1968	22														22	
1969	22														22	
1970	15														15	
1971	14														14	
1972	12						18		32						12	
1973	17	1.25					22	2.73	35	`					17	1.25
1974	16	1.26					23	1.35	55	1.22					16	1.26
1975	16	1.26	26	1.24			28	1.22	63	1.20					16	1.26
1976	18	1.23	23	1.26			22	1.36	54	1.19					18	1.23
1977	21	1.20	68	1.06			23	1.19	58	1.18					21	1.20
1978	23	1.19	133	1.16			22	1.19	52	1.20					23	1.19
1979	23	1.19	38	1.17			36	1.18	60	1.21					23	1.19
1980	22	1.19	20	1.30			18	1.23	38	1.17					22	1.19
1981	23	1.49	27	1.23	21	1.20	22	1.19	39	1.16					23	1.49
1982	18	1.23	23	1.19	20	1.21	29	1.21	45	1.19					18	1.23
1983	21	1.20	22	1.19	18	1.23	20	1.30	39	1.21					21	1.20
1984	20	1.21	65	1.19	19	1.41	22	1.19	43	1.20	19	1.22	23	1.19	20	1.21
1985	19	1.22	58	1.18	19	1.22	21	1.20	36	1.23	21	1.20	23	1.26	19	1.22
1986	19	1.22	29	1.21	24	1.26	21	1.20	40	1.21	22	1.19	23	1.19	19	1.22
1987	31	1.14	45	1.19	22	1.19	22	1.19	36	1.18	18	1.23	21	1.20	31	1.14
1988	21	1.47	100	1.41	22	1.45	19	1.41	36	1.44	20	1.40	23	1.44	21	1.47
1989	21	1.47	125	1.42	22	1.53	18	1.43	48	1.41	21	1.38	23	1.44	21	1.47
1990	24	1.43	116	1.42	15	1.43	14	1.41	28	1.42	15	1.43	16	1.43	24	1.43
1991	33	1.42	97	1.43	14	1.42	14	1.42	18	1.41	15	1.43	15	1.42	33	1.42
1992	14	1.11	112	1.11	13	1.11	13	1.11	17	1.11	13	1.11	13	1.11	14	1.11
1993	23	1.11	105	1.11	18	1.11	18	1.11	21	1.11	17	1.11	20	1.11	23	1.11
1994	21	1.11	62	1.11	19	1.11	18	1.11	21	1.11	18	1.11	20	1.11	21	1.11
1995	24	1.11	63	1.11	20	1.11	20	1.11	23	1.11	19	1.11	22	1.11	24	1.11
1996	23	1.11	57	1.11	19	1.11	19	1.11	20	1.11	19	1.11	22	1.11	23	1.11
1997	18	1.11	45	1.11	18	1.11	19	1.11	21	1.11	18	1.11	22	1.11	18	1.11
1998	17	1.11	45	1.11	22	1.11	22	1.11	23	1.11	21	1.11	23	1.11	17	1.11
1999	20	1.11													20	1.11
2000	21	1.11													21	1.11
2001	19	1.11													19	1.11

(a) Source: Plant perimeter values – SRS Annual Environmental Reports (see reference section).

C.2.2 Releases of Noble Gases

Argon-41 was released in large quantities during reactor operation. The potential dose resulting from submersion in this noble gas is external because the gas is not absorbed by the body. Calculated air concentrations were estimated for populated areas at SRS and then converted to organ and whole body doses. The annual doses (assuming 2,000 hours per year of exposure) have been tabulated based on 50th percentile source term and presented in Tables C-20 and C-21. The geometric standard deviations provided in these tables were based on the estimated 95th percentile source terms using the relationship provided in Section 3.1.1.

Table C-20 Argon-41 50th Percentile Dose[(a)] to Skin[(b)] and Bone Surfaces[(c)] Resulting From Reactor Releases (mrem per year) and geometric standard deviation

Year	A Area	GSD	C Area	GSD	D Area	GSD	F- Area	GSD	BG	GSD	H Area	GSD	S Area	GSD	Z Area	GSD	R Area	GSD
1955	0.17	1.15	1.81	1.15	0.30	1.15	0.39	1.15	0.42	1.15	0.42	1.15	0.36	1.15	0.35	1.15	2.14	1.15
1956	0.25	1.15	4.08	1.15	0.48	1.15	0.67	1.15	0.68	1.15	0.64	1.15	0.54	1.15	0.51	1.15	2.61	1.15
1957	0.31	1.15	4.22	1.15	0.56	1.15	0.77	1.15	0.79	1.15	0.78	1.15	0.66	1.15	0.63	1.15	4.09	1.15
1958	0.36	1.15	5.04	1.15	0.66	1.15	0.91	1.15	0.94	1.15	0.92	1.15	0.78	1.15	0.75	1.15	4.90	1.15
1959	0.46	1.15	6.67	1.15	0.85	1.15	1.17	1.15	1.21	1.15	1.17	1.15	0.99	1.15	0.94	1.15	5.80	1.15
1960	0.45	1.15	6.61	1.15	0.81	1.15	1.15	1.15	1.19	1.15	1.18	1.15	1.00	1.15	0.96	1.15	7.12	1.15
1961	0.45	1.15	6.61	1.15	0.81	1.15	1.15	1.15	1.19	1.15	1.16	1.15	0.98	1.15	0.94	1.15	6.46	1.15
1962	0.48	1.15	7.33	1.15	0.85	1.15	1.23	1.15	1.27	1.15	1.24	1.15	1.05	1.15	1.00	1.15	7.18	1.15
1963	0.50	1.15	7.41	1.15	0.90	1.15	1.27	1.15	1.31	1.15	1.28	1.15	1.08	1.15	1.04	1.15	7.22	1.15
1964	0.42	1.15	6.50	1.15	0.75	1.15	1.09	1.15	1.13	1.15	1.11	1.15	0.94	1.15	0.90	1.15	7.04	1.15
1965	0.39	1.15	5.76	1.15	0.70	1.15	0.99	1.15	1.03	1.15	1.01	1.15	0.86	1.15	0.82	1.15	6.28	1.15
1966	0.31	1.15	5.62	1.15	0.63	1.15	0.84	1.15	0.84	1.15	0.74	1.15	0.61	1.15	0.56	1.15	0.72	1.15
1967	0.36	1.15	6.43	1.15	0.72	1.15	0.96	1.15	0.96	1.15	0.85	1.15	0.70	1.15	0.64	1.15	0.82	1.15
1968	0.26	1.15	6.11	1.15	0.53	1.15	0.77	1.15	0.74	1.15	0.62	1.15	0.51	1.15	0.46	1.15	0.57	1.15
1969	0.17	1.15	3.84	1.15	0.34	1.15	0.49	1.15	0.47	1.15	0.39	1.15	0.32	1.15	0.29	1.15	0.35	1.15
1970	0.13	1.15	2.83	1.15	0.26	1.15	0.37	1.15	0.36	1.15	0.31	1.15	0.25	1.15	0.23	1.15	0.29	1.15
1971	0.16	1.15	3.27	1.15	0.31	1.15	0.45	1.15	0.44	1.15	0.38	1.15	0.32	1.15	0.29	1.15	0.39	1.15
1972	0.19	1.15	4.11	1.15	0.40	1.15	0.55	1.15	0.53	1.15	0.46	1.15	0.38	1.15	0.34	1.15	0.44	1.15
1973	0.21	1.15	3.60	1.15	0.43	1.15	0.55	1.15	0.54	1.15	0.48	1.15	0.40	1.15	0.37	1.15	0.50	1.15
1974	0.12	1.15	2.05	1.15	0.23	1.15	0.31	1.15	0.31	1.15	0.28	1.15	0.23	1.15	0.22	1.15	0.32	1.15
1975	0.08	1.15	1.81	1.15	0.15	1.15	0.23	1.15	0.22	1.15	0.18	1.15	0.15	1.15	0.14	1.15	0.17	1.15
1976	0.10	1.15	2.43	1.15	0.20	1.15	0.30	1.15	0.29	1.15	0.24	1.15	0.19	1.15	0.18	1.15	0.22	1.15
1977	0.08	1.15	2.28	1.15	0.16	1.15	0.25	1.15	0.24	1.15	0.20	1.15	0.16	1.15	0.14	1.15	0.16	1.15
1978	0.06	1.15	1.33	1.15	0.13	1.15	0.18	1.15	0.17	1.15	0.15	1.15	0.12	1.15	0.11	1.15	0.14	1.15
1979	0.06	1.15	1.58	1.15	0.13	1.15	0.19	1.15	0.18	1.15	0.15	1.15	0.13	1.15	0.11	1.15	0.14	1.15
1980	0.09	1.15	2.35	1.15	0.16	1.15	0.27	1.15	0.25	1.15	0.21	1.15	0.17	1.15	0.15	1.15	0.18	1.15
1981	0.07	1.15	1.72	1.15	0.15	1.15	0.22	1.15	0.21	1.15	0.18	1.15	0.14	1.15	0.13	1.15	0.16	1.15
1982	0.07	1.15	1.50	1.15	0.14	1.15	0.20	1.15	0.19	1.15	0.17	1.15	0.14	1.15	0.12	1.15	0.17	1.15
1983	0.04	1.15	0.50	1.15	0.08	1.15	0.10	1.15	0.10	1.15	0.10	1.15	0.08	1.15	0.08	1.15	0.13	1.15
1984	0.04	1.15	0.54	1.15	0.07	1.15	0.09	1.15	0.10	1.15	0.09	1.15	0.08	1.15	0.07	1.15	0.12	1.15
1985	0.05	1.15	0.84	1.15	0.10	1.15	0.14	1.15	0.14	1.15	0.13	1.15	0.11	1.15	0.10	1.15	0.17	1.15
1986	0.06	1.15	0.19	1.15	0.12	1.15	0.13	1.15	0.14	1.15	0.15	1.15	0.13	1.15	0.13	1.15	0.24	1.15
1987	0.09	0.86	0.33	0.97	0.21	0.98	0.19	0.89	0.20	0.87	0.21	0.83	0.17	0.82	0.17	0.81	0.26	0.66
1988	0.03	0.79	0.11	0.91	0.07	0.91	0.07	0.82	0.08	0.80	0.08	0.76	0.07	0.75	0.06	0.74	0.11	0.60
1989	0.00	1.15	0.00	1.15	0.00	1.15	0.00	1.15	0.00	1.15	0.00	1.15	0.00	1.15	0.00	1.15	0.00	1.15

(a) Doses estimated based on 2,000 hours per year exposure and FGR #12 dose conversion factor of 1.01E-13 Sv m^3 per sec Bq.

(b) Dose to skin should be assessed as 30% from >15 KeV electrons and 70% from >250 KeV photons.

(c) Dose to bone surfaces should be assumed to be 100% from >250 KeV photons.

Table C-20 (cont.) Argon-41 50th Percentile Dose[(a)] to Skin[(b)] and Bone Surfaces[(c)] Resulting From Reactor Releases (mrem per year) and geometric standard deviation

Year	RR Yard	GSD	P Area	GSD	L Area	GSD	K-Area	GSD	FS	GSD	CS	GSD	Ave-rage	GSD	Max-imum	GSD
1955	0.67	1.15	4.21	1.15	2.67	1.15	2.56	1.15	0.18	1.15	0.56	1.15	1.15	1.15	4.21	1.15
1956	0.78	1.15	4.09	1.15	4.27	1.15	4.23	1.15	0.26	1.15	1.02	1.15	1.67	1.15	4.27	1.15
1957	0.96	1.15	4.95	1.15	5.13	1.15	5.04	1.15	0.31	1.15	1.14	1.15	2.02	1.15	5.13	1.15
1958	1.13	1.15	5.78	1.15	5.99	1.15	5.90	1.15	0.37	1.15	1.35	1.15	2.39	1.15	5.99	1.15
1959	1.44	1.15	7.41	1.15	7.69	1.15	7.58	1.15	0.47	1.15	1.75	1.15	3.04	1.15	7.69	1.15
1960	1.38	1.15	6.72	1.15	6.93	1.15	6.84	1.15	0.47	1.15	1.72	1.15	2.97	1.15	7.12	1.15
1961	1.43	1.15	7.38	1.15	6.96	1.15	6.84	1.15	0.47	1.15	1.72	1.15	2.97	1.15	7.38	1.15
1962	1.48	1.15	7.44	1.15	7.01	1.15	6.91	1.15	0.49	1.15	1.85	1.15	3.12	1.15	7.44	1.15
1963	1.52	1.15	7.52	1.15	7.77	1.15	7.67	1.15	0.51	1.15	1.91	1.15	3.26	1.15	7.77	1.15
1964	1.26	1.15	5.95	1.15	6.12	1.15	6.05	1.15	0.44	1.15	1.64	1.15	2.76	1.15	7.04	1.15
1965	1.14	1.15	5.19	1.15	6.02	1.15	5.96	1.15	0.40	1.15	1.49	1.15	2.54	1.15	6.28	1.15
1966	0.94	1.15	5.56	1.15	5.88	1.15	5.84	1.15	0.29	1.15	1.34	1.15	2.05	1.15	5.88	1.15
1967	1.08	1.15	6.36	1.15	6.72	1.15	6.68	1.15	0.34	1.15	1.54	1.15	2.34	1.15	6.72	1.15
1968	0.77	1.15	5.23	1.15	1.01	1.15	5.49	1.15	0.24	1.15	1.28	1.15	1.64	1.15	6.11	1.15
1969	0.46	1.15	3.02	1.15	0.65	1.15	3.86	1.15	0.15	1.15	0.81	1.15	1.04	1.15	3.86	1.15
1970	0.40	1.15	2.73	1.15	0.51	1.15	2.85	1.15	0.12	1.15	0.61	1.15	0.82	1.15	2.85	1.15
1971	0.58	1.15	4.31	1.15	0.66	1.15	3.16	1.15	0.15	1.15	0.72	1.15	1.04	1.15	4.31	1.15
1972	0.60	1.15	4.18	1.15	0.79	1.15	4.45	1.15	0.18	1.15	0.90	1.15	1.23	1.15	4.45	1.15
1973	0.72	1.15	5.13	1.15	0.90	1.15	5.34	1.15	0.19	1.15	0.86	1.15	1.35	1.15	5.34	1.15
1974	0.49	1.15	3.77	1.15	0.52	1.15	2.36	1.15	0.11	1.15	0.49	1.15	0.79	1.15	3.77	1.15
1975	0.24	1.15	1.70	1.15	0.30	1.15	1.44	1.15	0.07	1.15	0.38	1.15	0.48	1.15	1.81	1.15
1976	0.29	1.15	1.99	1.15	0.37	1.15	1.90	1.15	0.09	1.15	0.50	1.15	0.62	1.15	2.43	1.15
1977	0.21	1.15	1.39	1.15	0.28	1.15	1.31	1.15	0.07	1.15	0.44	1.15	0.49	1.15	2.28	1.15
1978	0.19	1.15	1.30	1.15	0.25	1.15	1.42	1.15	0.06	1.15	0.29	1.15	0.39	1.15	1.42	1.15
1979	0.18	1.15	1.24	1.15	0.24	1.15	1.24	1.15	0.06	1.15	0.32	1.15	0.40	1.15	1.58	1.15
1980	0.24	1.15	1.67	1.15	0.30	1.15	1.31	1.15	0.08	1.15	0.46	1.15	0.53	1.15	2.35	1.15
1981	0.22	1.15	1.47	1.15	0.28	1.15	1.53	1.15	0.07	1.15	0.36	1.15	0.46	1.15	1.72	1.15
1982	0.24	1.15	1.76	1.15	0.28	1.15	1.33	1.15	0.07	1.15	0.32	1.15	0.45	1.15	1.76	1.15
1983	0.21	1.15	1.76	1.15	0.20	1.15	0.83	1.15	0.04	1.15	0.14	1.15	0.29	1.15	1.76	1.15
1984	0.20	1.15	1.67	1.15	0.17	1.15	0.50	1.15	0.04	1.15	0.14	1.15	0.26	1.15	1.67	1.15
1985	0.27	1.15	2.23	1.15	0.25	1.15	0.84	1.15	0.05	1.15	0.21	1.15	0.38	1.15	2.23	1.15
1986	0.44	1.15	3.76	1.15	0.36	1.15	1.27	1.15	0.07	1.15	0.15	1.15	0.49	1.15	3.76	1.15
1987	0.40	0.50	2.89	0.20	1.40	1.04	3.03	1.12	0.09	0.82	0.23	0.90	0.66	0.83	3.03	1.12
1988	0.18	0.45	1.31	0.18	0.71	1.05	0.80	1.11	0.03	0.75	0.09	0.83	0.25	0.75	1.31	0.82
1989	0.00	1.15	0.01	1.15	0.11	1.15	0.01	1.15	0.00	1.15	0.00	1.15	0.01	1.15	0.11	1.15

(a) Doses estimated based on 2,000 hours per year exposure and FGR #12 dose conversion factor of 1.01E-13 Sv m^3 per sec Bq.
(b) Dose to skin should be assessed as 30% from >15 KeV electrons and 70% from >250 KeV photons.
(c) Dose to bone surfaces should be assumed to be 100% from >250 KeV photons.

Table C-21 Argon-41 50th Percentile Dose[(a) (b)] to Whole Body and Other Organs Resulting From Reactor Releases (mrem per year) and geometric standard deviation

Year	A Area	GSD	C Area	GSD	D Area	GSD	F- Area	GSD	BG	GSD	H Area	GSD	S Area	GSD	Z Area	GSD	R Area	GSD
1955	0.12	1.15	1.28	1.15	0.21	1.15	0.28	1.15	0.30	1.15	0.30	1.15	0.26	1.15	0.25	1.15	1.52	1.15
1956	0.18	1.15	2.89	1.15	0.34	1.15	0.47	1.15	0.48	1.15	0.45	1.15	0.38	1.15	0.36	1.15	1.85	1.15
1957	0.22	1.15	2.99	1.15	0.40	1.15	0.54	1.15	0.56	1.15	0.55	1.15	0.47	1.15	0.45	1.15	2.90	1.15
1958	0.26	1.15	3.58	1.15	0.47	1.15	0.64	1.15	0.67	1.15	0.65	1.15	0.55	1.15	0.53	1.15	3.47	1.15
1959	0.33	1.15	4.73	1.15	0.61	1.15	0.83	1.15	0.86	1.15	0.83	1.15	0.70	1.15	0.67	1.15	4.11	1.15
1960	0.32	1.15	4.68	1.15	0.58	1.15	0.82	1.15	0.85	1.15	0.84	1.15	0.71	1.15	0.68	1.15	5.05	1.15
1961	0.32	1.15	4.68	1.15	0.58	1.15	0.81	1.15	0.84	1.15	0.83	1.15	0.70	1.15	0.67	1.15	4.58	1.15
1962	0.34	1.15	5.20	1.15	0.60	1.15	0.87	1.15	0.90	1.15	0.88	1.15	0.74	1.15	0.71	1.15	5.09	1.15
1963	0.35	1.15	5.25	1.15	0.64	1.15	0.90	1.15	0.93	1.15	0.91	1.15	0.77	1.15	0.74	1.15	5.12	1.15
1964	0.30	1.15	4.61	1.15	0.53	1.15	0.77	1.15	0.80	1.15	0.79	1.15	0.67	1.15	0.64	1.15	4.99	1.15
1965	0.28	1.15	4.09	1.15	0.50	1.15	0.71	1.15	0.73	1.15	0.72	1.15	0.61	1.15	0.58	1.15	4.45	1.15
1966	0.22	1.15	3.99	1.15	0.45	1.15	0.60	1.15	0.59	1.15	0.52	1.15	0.43	1.15	0.40	1.15	0.51	1.15
1967	0.25	1.15	4.56	1.15	0.51	1.15	0.68	1.15	0.68	1.15	0.60	1.15	0.49	1.15	0.45	1.15	0.58	1.15
1968	0.19	1.15	4.33	1.15	0.37	1.15	0.54	1.15	0.52	1.15	0.44	1.15	0.36	1.15	0.33	1.15	0.40	1.15
1969	0.12	1.15	2.72	1.15	0.24	1.15	0.35	1.15	0.33	1.15	0.28	1.15	0.23	1.15	0.21	1.15	0.25	1.15
1970	0.09	1.15	2.00	1.15	0.19	1.15	0.26	1.15	0.25	1.15	0.22	1.15	0.18	1.15	0.16	1.15	0.20	1.15
1971	0.11	1.15	2.32	1.15	0.22	1.15	0.32	1.15	0.31	1.15	0.27	1.15	0.22	1.15	0.21	1.15	0.28	1.15
1972	0.14	1.15	2.91	1.15	0.28	1.15	0.39	1.15	0.38	1.15	0.32	1.15	0.27	1.15	0.24	1.15	0.31	1.15
1973	0.15	1.15	2.55	1.15	0.31	1.15	0.39	1.15	0.39	1.15	0.34	1.15	0.28	1.15	0.26	1.15	0.35	1.15
1974	0.08	1.15	1.45	1.15	0.16	1.15	0.22	1.15	0.22	1.15	0.20	1.15	0.17	1.15	0.15	1.15	0.22	1.15
1975	0.05	1.15	1.29	1.15	0.11	1.15	0.16	1.15	0.15	1.15	0.13	1.15	0.11	1.15	0.10	1.15	0.12	1.15
1976	0.07	1.15	1.73	1.15	0.14	1.15	0.21	1.15	0.20	1.15	0.17	1.15	0.14	1.15	0.12	1.15	0.15	1.15
1977	0.06	1.15	1.62	1.15	0.11	1.15	0.18	1.15	0.17	1.15	0.14	1.15	0.11	1.15	0.10	1.15	0.12	1.15
1978	0.04	1.15	0.95	1.15	0.09	1.15	0.13	1.15	0.12	1.15	0.10	1.15	0.09	1.15	0.08	1.15	0.10	1.15
1979	0.05	1.15	1.12	1.15	0.09	1.15	0.14	1.15	0.13	1.15	0.11	1.15	0.09	1.15	0.08	1.15	0.10	1.15
1980	0.06	1.15	1.66	1.15	0.12	1.15	0.19	1.15	0.18	1.15	0.15	1.15	0.12	1.15	0.11	1.15	0.13	1.15
1981	0.05	1.15	1.22	1.15	0.10	1.15	0.15	1.15	0.15	1.15	0.12	1.15	0.10	1.15	0.09	1.15	0.11	1.15
1982	0.05	1.15	1.06	1.15	0.10	1.15	0.14	1.15	0.14	1.15	0.12	1.15	0.10	1.15	0.09	1.15	0.12	1.15
1983	0.03	1.15	0.35	1.15	0.06	1.15	0.07	1.15	0.07	1.15	0.07	1.15	0.06	1.15	0.06	1.15	0.09	1.15
1984	0.03	1.15	0.39	1.15	0.05	1.15	0.07	1.15	0.07	1.15	0.06	1.15	0.05	1.15	0.05	1.15	0.08	1.15
1985	0.04	1.15	0.60	1.15	0.07	1.15	0.10	1.15	0.10	1.15	0.09	1.15	0.08	1.15	0.07	1.15	0.12	1.15
1986	0.04	1.15	0.14	1.15	0.08	1.15	0.09	1.15	0.10	1.15	0.11	1.15	0.09	1.15	0.09	1.15	0.17	1.15
1987	0.06	0.86	0.24	0.97	0.15	0.98	0.13	0.89	0.14	0.87	0.15	0.83	0.12	0.82	0.12	0.81	0.19	0.66
1988	0.02	0.79	0.08	0.91	0.05	0.91	0.05	0.82	0.05	0.80	0.06	0.76	0.05	0.75	0.05	0.74	0.08	0.60
1989	0.00	1.15	0.00	1.15	0.00	1.15	0.00	1.15	0.00	1.15	0.00	1.15	0.00	1.15	0.00	1.15	0.00	1.15

(a) Doses estimated based on 2,000 hours per year exposure and FGR #12 dose conversion factor of 7.16E-14 Sv m^3 per sec Bq.

(b) Dose to whole body and other underlying organs should be assumed to be 100% from >250 KeV photons.

Table C-21 (cont.) Argon-41 95th Percentile Dose (a) (b) to Whole Body and Other Organs Resulting From Reactor Releases (mrem per year) and geometric standard deviation

Year	RR Yard	GSD	P Area	GSD	L Area	GSD	K-Area	GSD	FS	GSD	CS	GSD	Ave-rage	GSD	Max-imum	GSD
1955	0.47	1.15	2.99	1.15	1.90	1.15	1.81	1.15	0.12	1.15	0.40	1.15	0.81	1.15	2.99	1.15
1956	0.55	1.15	2.90	1.15	3.03	1.15	3.00	1.15	0.18	1.15	0.72	1.15	1.19	1.15	3.03	1.15
1957	0.68	1.15	3.51	1.15	3.63	1.15	3.58	1.15	0.22	1.15	0.81	1.15	1.43	1.15	3.63	1.15
1958	0.80	1.15	4.10	1.15	4.25	1.15	4.18	1.15	0.26	1.15	0.96	1.15	1.69	1.15	4.25	1.15
1959	1.02	1.15	5.26	1.15	5.45	1.15	5.38	1.15	0.33	1.15	1.24	1.15	2.16	1.15	5.45	1.15
1960	0.98	1.15	4.76	1.15	4.91	1.15	4.85	1.15	0.33	1.15	1.22	1.15	2.10	1.15	5.05	1.15
1961	1.01	1.15	5.23	1.15	4.93	1.15	4.85	1.15	0.33	1.15	1.22	1.15	2.11	1.15	5.23	1.15
1962	1.05	1.15	5.28	1.15	4.97	1.15	4.90	1.15	0.35	1.15	1.31	1.15	2.21	1.15	5.28	1.15
1963	1.08	1.15	5.33	1.15	5.51	1.15	5.44	1.15	0.36	1.15	1.35	1.15	2.31	1.15	5.51	1.15
1964	0.90	1.15	4.22	1.15	4.34	1.15	4.29	1.15	0.31	1.15	1.16	1.15	1.96	1.15	4.99	1.15
1965	0.81	1.15	3.68	1.15	4.27	1.15	4.23	1.15	0.29	1.15	1.05	1.15	1.80	1.15	4.45	1.15
1966	0.67	1.15	3.94	1.15	4.17	1.15	4.14	1.15	0.21	1.15	0.95	1.15	1.45	1.15	4.17	1.15
1967	0.76	1.15	4.51	1.15	4.76	1.15	4.73	1.15	0.24	1.15	1.09	1.15	1.66	1.15	4.76	1.15
1968	0.54	1.15	3.71	1.15	0.72	1.15	3.89	1.15	0.17	1.15	0.91	1.15	1.16	1.15	4.33	1.15
1969	0.33	1.15	2.14	1.15	0.46	1.15	2.73	1.15	0.11	1.15	0.57	1.15	0.74	1.15	2.73	1.15
1970	0.28	1.15	1.94	1.15	0.36	1.15	2.02	1.15	0.08	1.15	0.43	1.15	0.58	1.15	2.02	1.15
1971	0.41	1.15	3.06	1.15	0.47	1.15	2.24	1.15	0.11	1.15	0.51	1.15	0.74	1.15	3.06	1.15
1972	0.43	1.15	2.96	1.15	0.56	1.15	3.16	1.15	0.13	1.15	0.64	1.15	0.87	1.15	3.16	1.15
1973	0.51	1.15	3.64	1.15	0.64	1.15	3.79	1.15	0.14	1.15	0.61	1.15	0.96	1.15	3.79	1.15
1974	0.35	1.15	2.67	1.15	0.37	1.15	1.67	1.15	0.08	1.15	0.35	1.15	0.56	1.15	2.67	1.15
1975	0.17	1.15	1.20	1.15	0.21	1.15	1.02	1.15	0.05	1.15	0.27	1.15	0.34	1.15	1.29	1.15
1976	0.21	1.15	1.41	1.15	0.27	1.15	1.35	1.15	0.06	1.15	0.35	1.15	0.44	1.15	1.73	1.15
1977	0.15	1.15	0.98	1.15	0.20	1.15	0.93	1.15	0.05	1.15	0.31	1.15	0.35	1.15	1.62	1.15
1978	0.13	1.15	0.92	1.15	0.18	1.15	1.00	1.15	0.04	1.15	0.20	1.15	0.28	1.15	1.00	1.15
1979	0.13	1.15	0.88	1.15	0.17	1.15	0.88	1.15	0.04	1.15	0.23	1.15	0.28	1.15	1.12	1.15
1980	0.17	1.15	1.19	1.15	0.21	1.15	0.93	1.15	0.06	1.15	0.33	1.15	0.37	1.15	1.66	1.15
1981	0.15	1.15	1.04	1.15	0.20	1.15	1.09	1.15	0.05	1.15	0.26	1.15	0.33	1.15	1.22	1.15
1982	0.17	1.15	1.25	1.15	0.20	1.15	0.94	1.15	0.05	1.15	0.23	1.15	0.32	1.15	1.25	1.15
1983	0.15	1.15	1.24	1.15	0.14	1.15	0.59	1.15	0.03	1.15	0.10	1.15	0.21	1.15	1.24	1.15
1984	0.14	1.15	1.18	1.15	0.12	1.15	0.35	1.15	0.03	1.15	0.10	1.15	0.18	1.15	1.18	1.15
1985	0.19	1.15	1.58	1.15	0.18	1.15	0.60	1.15	0.04	1.15	0.15	1.15	0.27	1.15	1.58	1.15
1986	0.31	1.15	2.66	1.15	0.26	1.15	0.90	1.15	0.05	1.15	0.11	1.15	0.35	1.15	2.66	1.15
1987	0.29	0.50	2.05	0.20	0.99	1.04	2.15	1.12	0.06	0.82	0.16	0.90	0.47	0.83	2.15	1.12
1988	0.12	0.45	0.93	0.18	0.50	1.05	0.57	1.11	0.02	0.75	0.06	0.83	0.18	0.75	0.93	0.82
1989	0.00	1.15	0.01	1.15	0.08	1.15	0.01	1.15	0.00	1.15	0.00	1.15	0.01	1.15	0.08	1.15

(a) Doses estimated based on 2,000 hours per year exposure and FGR #12 dose conversion factor of 7.16E-14 Sv m^3 per sec Bq.

(b) Dose to whole body and other underlying organs should be assumed to be 100% from >250 KeV photons.

ATTACHMENT D
OCCUPATIONAL INTERNAL DOSE FOR MONITORED WORKERS

D.0 Occupational Internal Dose

The bioassay program monitored internal uptake of radionuclides using both *in vitro* analysis of urine and feces and *in vivo* monitoring using whole body and chest counting.

D.1 *In vitro* Minimum Detectable Activities (MDAs) and Reporting Levels

The *decision level* is the level at which activity is considered present in a sample with a 95% confidence level. The *reporting level* is the minimally acceptable decision level. It appears from the database that at some times and certain radionuclides the reporting level was also the level below which a result was reported as "less than", regardless of the actual decision level for a given sample. Since 1990, the MDA has been defined as the quantity of material (or activity) that has a 5% chance of not being detected. Prior to 1990, the term MDA may have had less rigorous or at least slightly different meanings. For some times and certain radionuclides, MDA and reporting level were synonymous. For the dosimetrist the reporting level is the relevant concept over the history of the SRS records. There is no consistent quantitative relationship between the reporting level and the MDA.

D.1.1 *In vitro* Urine Analysis

The reporting levels and MDAs (if known and different than reporting levels), are shown in Table D-01. These values are applicable over the time period specified. The lack of an MDA in a given time period reflects the lack of documented information about urine analysis for that radionuclide. Activity fractions for depleted, recycled, and high enriched uranium are listed in the glossary section. Activity fractions for 6% and 12% plutonium mixtures are given in Chapter 4.

The following codes are found in the excreta records:

C routine urine sample, includes baselines and termination samples
A-A 24-hours special urine sample
A-O two 24-hour urine samples, one analyzed, one held
A-W follow-up to positive routine sample
IA Inconclusive Analysis; analysis didn't meet QC criteria; no result reported
B follow-up to an IA sample
A-F single void fecal sample
T tritium sample, only 50 mL collected
P-10 an old code for tritium
SP Neptunium
FP fission products; gamma scan.
FPIA fission products/induced activity. Same analysis as FP.
IA with a numerical result would be same as FP
LMF uranium

D.1.2 *In vitro* Urine Analyses for Individual Radionuclides

Urine analysis for trivalent actinides (americium, curium, and californium)

Urinalysis for trivalent actinides began in the mid-1960s. From initiation to 1990, the reporting levels varied from about 1 dpm/1.5L to 0.3 dpm/L. In 1990, the reporting levels decreased to 0.1 dpm/L. The MDAs since 1994 are shown in Table D-01. The counting method does not distinguish ^{241}Am from ^{244}Cm from ^{252}Cf. Interpretation of which radionuclide is being counted will have to come from particulars of the intake description. The claimant-favorable default is ^{241}Am.

Urine analysis for plutonium

The reporting levels and MDAs are also listed in Table D-01. Results prior to 1981 (1988 for special samples) were for total alpha from plutonium isotopes. Plutonium-239 results are usually indicative of a mixture of plutonium isotopes. See 4.1.2 for discussion.

Urine analysis for tritium

From startup until 1958, the reported MDA was 1 µCi/L. From 1958 until present, the MDA improved to the current level of 20 nCi/L; however, the reporting level remained at 1 µCi/ L, and decreased to the current value of 0.1 µCi/L. For tritium, the denominator has always been per *liter* of urine, and not 1.5 L as for other radionuclides.

For current analyses, tritium results of ≤0.05 µCi/L are reported as <0.1µCi/L, from 0.05 to 0.1 µCi/L are reported as 0.1 µCi/L, and >0.1 µCi/ L are reported as measured. Only results greater than 5 µCi/L were evaluated prior to computer evaluation of data.

Urine analysis for uranium

A variety of methods has been used historically to analyze uranium resulting in differences in the MDA over time, as shown in Table D-01. Activity fractions for depleted, recycled, and highly enriched uranium are listed in the glossary section. For earlier monitoring periods, the designations “enriched” and “depleted” analysis for uranium referred to the analysis and were not indicative of the degree of uranium enrichment. Natural background levels of uranium in excreta samples can interfere with interpretation of bioassay results; see the discussion in 4.1.2.

Urine analysis for radiostrontium

The start date for urinalysis for radioisotopes of strontium was not specifically determined, other than “late 1950s.” This was referred to as the fission product analysis. From start until 1969 the same method was used, which involved separation of the strontium and beta counting. Both ^{89}Sr and ^{90}Sr would have been counted, but it is claimant-favorable to assume the result is all ^{90}Sr.

D.1.3 *In vitro* Fecal Analysis

Fecal analysis has been done periodically since the 1950s for special purposes and was always done in conjunction with urine bioassay sampling. Usually the analysis was a gamma scan on the total sample with typical MDAs of 7 pCi for ^{241}Am, 300 pCi for ^{238}Pu, and 600 pCi for ^{238}Pu plus ^{239}Pu in weapons grade mixtures. If fecal samples were also analyzed by wet radiochemistry, the MDAs were approximately 1.5 times the same MDAs for urinalyses.

Table D-01 Limits of Detection for Urinalysis[a]

Radionuclide	Time Period	MDA (pCi/L)	Reporting Level (pCi/L)
^{3}H	Start to 1958		1 E+06
^{3}H	1958 to 2002		5 E+05
^{3}H	Present	2.0E+04	1E+05[b]
Sr	Start to 1969 (see fission product)		
^{90}Sr	Present	2.9	4
FP	1955 to 1960		30 dpm/75mL
FP	1960-1965		100 dpm/1.5L (beta) 500 dpm/1.5L (gamma)
FP	1965 - ?		1 nCi/1.5L
^{234}U	1990 to present	0.032	
^{235}U	1982 to 1986	0.14 ng	NA
^{235}U	1990 to present	0.036	
^{238}U	1990 to present	0.032	
U-natural	1982 to 1986	1 μg/L	1 μg/L
DU	1954 to 1982	Not listed	Not listed
DU	1982 to 1986	same as U-nat	same as U-nat
DU	1986-1994	0.5/1.5L	1 dpm/1.5L
RU	1990 to present	use ^{234}U and isotopic ratios	use ^{234}U and isotopic ratios
EU	Start to mid-1960s		0.15 dpm/1.5 L
EU	mid-1960s to 1982		1 dpm/1.5 L
EU	1982 to 1986		1 dpm/L
EU	1986 to 1990		1dpm/1.5 L
HEU	1990 to present	use ^{235}U and isotopic ratios	use ^{235}U and isotopic ratios
SP (^{237}Np)	~1959 to 1964		0.05 dpm/1.5L
SP	1964 to 2001		0.1 dpm/1.5L
^{237}Np	2001 to present	0.035 [c]	
^{238,239,240}Pu	1954 to 1962		0.05dpm/1.5L
^{238,239,240}Pu	1963 to 1981		0.1dpm/1.5 L
^{238}Pu	1981 to 1988[d]		0.05/1.5 L
^{238}Pu	1988 to present		0.07
^{239}Pu, ^{240}Pu	1981 to 1988[d]		0.07/1.5 L

Radionuclide	Time Period	MDA (pCi/L)	Reporting Level (pCi/L)
^{239}Pu, ^{240}Pu	1988 to present		0.06
^{241}Am	Mid 1960s to1971		3 dpm/1.5 L
^{241}Am	1971 to 1990		0.3 dpm/1.5 L
^{241}Am	1990 to 1994		0.1 dpm/L
^{241}Am	1994 to present	0.029	
^{244}Cm	Mid 1960s to1971		3 dpm/1.5 L
^{244}Cm	1971 to 1990		0.3 dpm/1.5 L
^{244}Cm	1990 to 1994		0.1 dpm/L
^{244}Cm	1994 to present	0.021	
^{252}Cf	Mid 1960s to1971		3 dpm/1.5 L
^{252}Cf	1971 to 1990		0.3 dpm/1.5 L
^{252}Cf	1990 to 1994		0.1 dpm/L
^{252}Cf	1994 to present	0.021	

(a) MDAs are given if less than reporting levels; generally the reporting level should be used as the limit of detection.
(b) Measured values between 5E+4 and 1E+5 pCi/L are reported in the database as 0.1 µCi/L.
(c) Some uncertainty about this value. See discussion in Chapter 4.
(d) For routine samples; gross alpha method still used for special samples.

D.2 *In vivo* MDAs and reporting practices at SRS

Whole body counting has been performed since 1960 and chest counting was initiated in the early 1970s. *In vivo* detection capabilities are reviewed below.

D.2.1 Whole Body Counting

A variety of hardware and software applications have been used for whole body counting since 1960 consistent with industry-wide improvements in the discipline. Whole body counting was not used for plutonium and americium bioassay due to their low energy photon emissions. Sometimes MDAs were reported on the forms for the individual measurements and these should be used when available. Default MDAs for various radionuclides are shown in Table D-02. MDAs and reporting levels were synonymous.

Table D-02 MDAs for Whole Body Counting

Radionuclide	Time Period	MDA (nCi)
^{54}Mn	Present	3.4
^{60}Co	Present	2.9
^{65}Zn	1960 to ??	5.1
^{65}Zn	Present	6.1
Zr/^{95}Nb	1960 to ??	2.2
^{106}Ru	1960 to ??	6.1
^{106}Ru	Present	36
^{125}Sb	Present	14
^{131}I	1960 to ??	1.4
^{134}Cs	Present	3.8
^{137}Cs	1960 to ??	1.0
^{137}Cs	Present	4.1
Ba/^{140}La	1960 to ??	9.3
^{144}Ce	1960 to ??	29
^{144}Ce	Present	69
^{152}Eu	Present	18
^{154}Eu	Present	8.4
^{235}U	Present	14
U-natural (^{235}U)	1960 to ??	62
^{237}Np	Present	14

For special investigational counts, the MDAs are approximately 0.71 times these values for whole body counting

D.2.2 Chest Counting

No MDA information could be found for chest counting performed from 1966 to around 1971. From 1971 to 1987 or 1988, using phoswich detectors, chest burdens of 4 to 10 nCi were reported for ^{238}Pu. In 1987 or 1988, the ^{238}Pu MDA is reported to be 60-70 nCi.

MDAs for later periods have been generated individually for counts by processing software, with typical values shown in Table D-03.

Table D-03 MDAs for Chest Counting

Radionuclide	Time Period	MDA (nCi)
^{144}Ce	Present	0.31
^{152}Eu	Present	0.056
^{228}Th	Present	3.2
^{232}Th	Present	31
^{234}U	Present	30
^{235}U	Present	0.10
^{236}U	Present	89
^{238}U	Present	1.1
DU	Present	1.2
RU	Present	8.3
HEU	Present	5.2
^{237}Np	Present	0.31
^{238}Pu	<1971	NA
^{238}Pu	1971 to 1987 or 1988	4 to 10
^{238}Pu	1987 to Present	130[a]
^{239}Pu	1987 to Present	290[a]
^{Pu}240	1987 to Present	110[a]
^{241}Am	1987 to Present	0.13[a]
^{243}Am	1987 to Present	0.12
^{244}Cm	1987 to Present	78[a]
^{252}Cf	1987 to Present	85[a]

Based on chest wall thickness of 50th male worker of 3.7 cm
For special investigational counts, the MDAs are approximately 0.77 times the values for chest counting in Table D-03.

ATTACHMENT E
OCCUPATIONAL EXTERNAL DOSE FOR MONITORED WORKERS

E.0 Occupational External Dose

The information needed to evaluate claims is directed to the technical parameters of the annual estimates of the primary organ dose that is calculated from the dosimeter interpreted personal dose equivalent, Hp(10), and, in the case of skin, testicular and breast cancer, Hp(0.07), used as a consistent basis of comparison for all years of SRS occupational radiation exposure.

The primary IREP screen used to input dose parameters is illustrated in Table E-01. Input to these fields is obtained from the SRS dose of record. The claim provides the primary organ of interest and other worker information needed to run IREP. Guidance to the dose reconstruction analyst to select technical external dose parameters to complete the respective fields in Table E-01 is presented in the following sections.

Table E-01 IREP Dose Parameter Input Screen

Exposure			Radiation Type	Distribution Parameters			
#	Year	Rate		Type	1	2	3
1	1960	Acute	Photon, 30-250 keV	Normal	2	2	0
2	1961	Acute					

E.1 Years of Exposure

The years of exposure should be identified from the SRS radiation dose reports. Missed dose is calculated for all zero or missing records within a reporting year as noted by non-zero or blank entries. SRS policies require monitoring of all workers occupationally exposed. Missed dose should be calculated for a claimant for each year of record as an employee unless there are valid reasons for years in which there is no SRS recorded dose.

E.2 Rate

As identified in OCAS-IG-001, acute is selected for all types of external beta and photon dose. Chronic is selected for tritium and neutron dose.

E.3 Radiation Type

E.3.1 Beta/Photon

Claimant-favorable assumptions should be made using guidance in Table E-02 for beta and photon (i.e., x-ray and gamma rays) radiation types to assure that dose is not underestimated. The values presented in this table are intended to provide a reasonable estimate of parameters used to calculate the organ dose without significant numerical error for long-term SRS workers in the respective facilities. There is no direct evidence to select the specific values shown other than considerations of the radiation sources and usual work tasks. In those cases where there is some doubt in the values, a range of

realistic values should be selected for comparison and the most claimant-favorable option selected.

Table E-02 Selection of Beta and Photon Radiation Energies and Percentages for SRS Facilities and Processes

Process/ Buildings	Description	Operations		Radiation Type	Energy Selection	Percent
		Begin	End			
Fuel Fabrication	Produced reactor fuel and target assemblies.	1953	1989			
300M	Uranium fuel fabrication	1953	1989	Beta Photon	> 15 keV 30 – 250 keV	100% 100%
Reactors	**During Operation:** Highly dispersed fields of higher-energy photon radiation fields from fission process, activation/ fission product nuclides. Potential for significant airborne nuclides and there may be significant higher-energy beta radiation.					
	While not in Operation: Highly dispersed fields of higher-energy photon radiation fields from activation/fission product nuclides. There may be significant higher-energy beta radiation during maintenance work resulting from fission products.			Beta	> 15 keV	100%
105C	Heavy Water Moderated Reactors	3/1955	6/1985	Photon	30 – 250 keV	50%
105K		10/1954	1988		> 250 keV	50%
		1992	7/1992			
105L		7/1954	1968			
		1985	6/1988			
105P		2/1954	8/1988			
105R		12/1953	6/1964			
Processing Plants	**Radiochemical Operations:** Highly dispersed fields of higher-energy photon radiation fields from activation/fission product nuclides dominant to most exposure profiles. Potential for higher-energy beta radiation during sampling and, maintenance work resulting from fission products.			Beta	> 15 keV	100%
200F	Radiochemical processing plant	1954		Photon	30 – 250 keV	50%
200H	Radiochemical processing plant	1955			> 250 keV	50%
Plutonium Production	**Plutonium Component Production**: Plutonium is machined into weapon components using a glove box assembly process with predominant close anterior exposure to workers. Radiation characteristics in this area involve significant lower-energy photons and lower-energy neutron radiation.			Photon	< 30 keV 30 – 250 keV	25% 75%
	Plutonium Storage: Radiation characteristics in this area generally involve dispersed lower-energy neutron radiation and scattered photons to include 60 keV Am-241 gamma ray.					
221F, B-line	Plutonium Finishing Process	1954				
221H, B-line	Plutonium Finishing Process	1955				
772F	Production Control Laboratory	1955				
235F	Plutonium Fuel Facility	1955				
736A	Calibration Facility	1953				
Radionuclide Production	Radiation characteristics are highly dependent upon the radionuclide being produced. SRS has produced several radionuclides including tritium, ^{238}Pu, ^{239}Pu, americium, curium, californium, etc.			(Specific to radio-nuclide) Photon, default	30 – 250 keV > 250 keV	50% 50%
773A	Savannah River Laboratory	1953				
Waste Handling	Radiation characteristics are highly dependent upon the source of waste but typically fission product nuclides (^{90}Sr/^{90}Y, ^{137}Cs) are dominant.			Beta	> 15 keV	100%
200F	General waste handling, storage tanks, etc.	1953	Ongoing	Photon	30 – 250 keV	50%
200H	General waste handling, storage tanks, etc	1953	Ongoing		> 250 keV	50%

E.3.2 Neutron

Table E-03 summarizes the default neutron dose distributions by energy for each neutron exposure area.

Table E-03 Selection of Neutron Radiation Energies and Percentages for SRS Facilities and Processes

Process	Description / Buildings	Operations Begin	Operations End	Neutron Energy	Default Dose Fraction (%)
Fuel Fabrication	Produced reactor fuel and target assemblies. Intermittent production mixed oxide fuel and plutonium – aluminum (Pu-Al) alloy targets.				
	(321-M)			0.1-2 MeV 2-20 MeV	83% 17%
Reactors	During Reactor Operation: Low-level neutron exposure through shielding. In addition, some intermittent neutron exposure from mixed oxide fuel and targets.				
	(105C)	3/1955	1/1987	10-100 keV 0.1 – 2 MeV	15% 85%
	(105K)	10/1954	1/1988		
		1992	7/1992		
	(105L)	7/1954	2/1968		
		1985	6/1988		
	(105P)	2/1954	8/1988		
	(105R)	12/1953	6/1964		
Plutonium Production	Plutonium Finishing Process: Plutonium enters the process as PuF_4 and is then fired into production pucks. Work is primarily conducted in glove boxes with predominant close anterior exposure to workers. Radiation levels at the beginning of the process are fairly constant while levels at end of process are closely related to production levels.				
	Plutonium Finishing Process (221-H, B-Line)	11/1954	1988	0.1-2 MeV 2-20 MeV	60% 40%
	Plutonium Finishing Process (221-F, B-Line)	10/1955	1988	0.1-2 MeV	100%
	Plutonium Storage			0.1-2 MeV 2-20 MeV	86% 14%
	Production Control Laboratory (772-F)	1955		0.1-2 meV	100%
	Plutonium Fuel Facility (235-F)	1955		0.1-2 meV	100%
Radionuclide Production and Calibration	Radiation characteristics are highly dependent upon the radionuclide being produced. SRS has produced several radionuclides including ^{238}Pu, ^{239}Pu, americium, curium, californium, etc. At the calibration facility radiological instruments are calibrated using a variety of sources with different neutron energy spectra.				
	Californium Production Facility (773-A)			0.1-2 MeV 2-20 MeV	50% 50%
	Calibration Facility (736-A)			0.1-2 MeV 2-20 MeV	83% 17%

E.4 Distribution Parameters

Selection of the distribution parameters in Table E-01 involves adjustments to the dose of record for missed dose prior to entry into the IREP input screen. The selection of a normal distribution for the type determines the definition of Parameters #1 and #2. For a normal distribution, Parameter #3 is not used.

E.4.1 Parameter #1

For a normal distribution, parameter #1 is the mean of the distribution of recorded dose for each year of monitoring. Prior to calculating this, it is necessary to adjust the recorded dose for exposure profiles described in Section E.4.1.1 where the recorded dose of record will under-estimate Hp(10).

E.4.1.1 Resolution of Photon, Neutron and Tritium Dose

For workers who terminated SRS employment prior to 1979, SRS hard-copy radiation records must be consulted to separate the photon, neutron and tritium dose for workers. Hard copy records do separate the recorded whole body dose (i.e., photon + neutron + tritium) with the exception of the period from the second quarter of 1963 through December 31, 1972. The missing dose components of the total whole body dose could be estimated from the average of the prior year (i.e., before second quarter of 1963) and subsequent year (after 1972) dose results according to the proportion of dose from photon, neutron and tritium dose based on examination of continuity in the worker's job and work activities. A claimant-favorable option is simply to assign the whole body dose (i.e., photon + neutron + tritium) as a photon dose, and using the methods in the following sections a claimant-favorable neutron dose will also be calculated assuming the worker was working in a facility with neutron exposure.

E.4.1.2 Adjustments to Recorded Photon Dose

Adjustments to the SRS reported photon dose are necessary to arrive at a claimant-favorable dose considering the uncertainty associated primarily with the complex workplace radiation fields and exposure orientations. Taylor et al (1995) identified calibration corrections of 1.119 and 1.039 to the reported SRS TLD-Deep photon dose for the periods prior to 1986 and 1987, respectively, to estimate the Hp(10) dose. This is summarized in Table E-04.

Table E-04 Adjustments to Reported SRS Deep Photon Dose

Time Period	Dosimeter	Facility	Step	Adjustment to Reported Dose
Prior to 1987	All beta/photon dosimeters	All facilities	A	Multiply reported TLD-Deep photon dose by a factor of 1.119 to estimate Hp(10).
For the year of 1987	TLD beta/photon dosimeter	All facilities	B	Multiply reported TLD-Deep photon dose by a factor of 1.039 to estimate Hp(10).
Post 1987	TLD beta/photon dosimeter	All facilities		No adjustments made.

E.4.1.3 Adjustments to Recorded Neutron Dose

The Savannah River Site (SRS) incorporated the energy variation of the dose equivalent into their calibration methodology. As a result, the recorded neutron dose is a combination of all neutron energies. In order to calculate the dose input to IREP (Table E-01), the recorded neutron dose must be separated into neutron energy groups as shown in Table E-03 and subsequently converted to ICRP 60 (1990) methodology.

The neutron dose equivalent correction factor can be calculated by dividing the dose fractions from Table E-04 for each facility and neutron energy category group shown in Table E-02 by multiplying the recorded neutron dose by the area specific correction factors the neutron dose equivalent is calculated. For example consider a 1000 mrem recorded neutron dose by an energy employee working at the HB-Line, the corrected neutron dose is 1144 mrem from neutrons between 0.1-2.0 MeV ***and*** 529 mrem from neutrons with energy between 2 and 20 MeV. Thus the corrected neutron dose is a total of 1673 mrem. These corrections should be applied to measured dose, missed dose and dose determined based on a neutron to photon ratio. Table E-05 summarizes the dose fractions and associated correction factors for each neutron exposure area.

E.4.1.4 Unmonitored Photon Dose

Adjustments to the recorded annual dose can be made using dose results for co-workers or using the recorded dose prior to and after the period of missed dose. These situations do require careful examination.

E.4.1.5 Missed Photon Dose

The missed photon dose for dosimeter results less-than the MDL is particularly important for earlier years when MDLs were higher and dosimeter exchange was more frequent. OCAS-IG-001 guidance should be followed to calculate the missed dose by using a claimant favorable maximum potential missed dose. This is calculated using the Minimum Detection Level (MDL) multiplied by the number of zero dose results to provide an estimate of the maximum potential missed dose. The following sections describe the potential missed photon dose corrections according to facility/location, dosimeter type, year and energy range.

Table E-05 Summary of neutron Dose Fractions and associated ICRP 60 correction factors

Process	Description / Buildings	Operations Begin	Operations End	Neutron Energy	Default Dose Fraction (%)	ICRP 60 Correction Factor
Fuel Fabrication	Produced reactor fuel and target assemblies. Intermittent production mixed oxide fuel and plutonium – aluminum (Pu-Al) alloy targets.					
	(321-M)			0.1-2 MeV 2-20 MeV	83% 17%	1.58 0.23
Reactors	During Reactor Operation: Low-level neutron exposure through shielding. In addition, some intermittent neutron exposure from mixed oxide fuel and targets.					
	(105C)	3/1955	1/1987	10-100 keV 0.1 – 2 MeV	15% 85%	0.28 1.62
	(105K)	10/1954	1/1988			
		1992	7/1992			
	(105L)	7/1954	2/1968			
		1985	6/1988			
	(105P)	2/1954	8/1988			
	(105R)	12/1953	6/1964			
Plutonium Production	Plutonium Finishing Process: Plutonium enters the process as PuF_4 and is then fired into production pucks. Work is primarily conducted in glove boxes with predominant close anterior exposure to workers. Radiation levels at the beginning of the process are fairly constant while levels at end of process are closely related to production levels.					
	Plutonium Finishing Process (221-H, B-Line)	11/1954	1988	0.1-2 MeV 2-20 MeV	60% 40%	1.14 0.53
	Plutonium Finishing Process (221-F, B-Line)	10/1955	1988	0.1-2 MeV	100%	1.91
	Plutonium Storage			0.1-2 MeV 2-20 MeV	86% 14%	1.64 0.19
	Production Control Laboratory (772-F)	1955				
	Plutonium Fuel Facility (235-F)	1955				
Radionuclide Production and Calibration	Radiation characteristics are highly dependent upon the radionuclide being produced. SRS has produced several radionuclides including ^{238}Pu, ^{239}Pu, americium, curium, californium, etc. At the calibration facility radiological instruments are calibrated using a variety of sources with different neutron energy spectra.					
	Californium Production Facility (773-A)			0.1-2 MeV 2-20 MeV	50% 50%	
	Calibration Facility (736-A)			0.1-2 MeV 2-20 MeV	83% 17%	

E.4.1.5.1 Facility/Location

The potential missed photon dose for the laboratory determined MDL and exchange frequency is presented in Table E-06.

Table E-06 Missed photon dose according to SRS Facility.

SRS Facility Type	Dosimeter	Period of Use	Exchange Frequency	MDL (mSv)[a]	Max. Annual Missed Dose (mSv)[b]
Fuel Fabrication, SRS Reactors, Processing Plants, Waste Handling	ORNL Two-Element film Dosimeter	Prior to 3/52	Weekly	0.4	20.8
	SRS Two Element film Dosimeter	After 3/52 through 9/30/1957	Weekly	0.4	20.8
		10/1/1957 through 11/8/1959	Biweekly	0.4	10.4
Above plus plutonium and nuclide production	SRS Multi-Element Film Dosimeter	11/9/1959 through 12/31/1964	Biweekly	0.4	10.4
		1/1/1965 through 12/31/1965	4-week	0.4	5.2
		1/1/1966 through 3/31/1970	Monthly	0.4	4.8
All SRS	SRS TLD	4/1/1970 through 6/30/1983	Monthly	0.15	1.8
	Panasonic TLD	7/1/1983 to 2003 (ongoing)	Monthly	0.05	0.6
	Panasonic TLND	1/1/1995 to 2003 (ongoing)	Monthly (n=12)	0.15	0.9

a. Estimated film dosimeter detection levels based on NIOSH (1993), NRC (1989), and Wilson et al (1990). TLD detection levels from Taylor et al (1995) and SRS (1993).
b. Maximum annual missed dose calculated from OCAS-IG-001 (NIOSH 2000).

E.4.1.5.3 Dosimeter Type

Table E-07 presents the maximum potential missed dose according to SRS facility, time period, exchange frequency and MDL.

Table E-07 Missed Photon Dose According to Dosimeter Type

Dosimeter Type	Period of Use	MDL (mSv)	Exchange Frequency	Max. Annual Missed Dose (mSv)
Two Element Film	Prior to 3/52 through 9/30/1957	0.4	Weekly (n = 52)	20.8
	10/1/1957 through 11/8/1959	0.4	Biweekly (n=26)	10.4
Multi-Element Film	11/9/1959 through 12/31/1964	0.4	Biweekly (n=26)	10.4
	1/1/1965 through 12/31/1965	0.4	4-week (n=13)	5.2
	1/1/1966 through 3/31/1970	0.4	Monthly (n=12)	4.8
TLD	4/1/1970 through 6/30/1983	0.15	Monthly (n=12)	1.8
	7/1/1983 to 2003 (ongoing)	0.05	Monthly (n=12)	0.6
		0.05	Quarterly (n = 4)	0.2

E.4.1.5.3 Year

Table E-08 presents the potential maximum missed photon dose according to year (actually by period according to dosimeter type and exchange).

Table E-08 Missed Photon Dose According to Year

Period of Use[(a)]	MDL (mSv)	Exchange Frequency	Max. Annual Missed Dose (mSv)
Prior to 1958	0.4	Weekly (n = 52)	20.8
1959	0.4	Biweekly (n=26)	10.4
1960 through 1964	0.4	Biweekly (n=26)	10.4
1965	0.4	4-week (n=13)	5.2
1966 through 1969	0.4	Monthly (n=12)	4.8
1970 through 1983	0.15	Monthly (n=12)	1.8
1984 to 2003 (ongoing)	0.05	Monthly (n=12)	0.6

(a) Actual periods by mm/dd/yyyy categorized according to year.

E.4.1.5.4 Energy Range

An estimate of the missed photon dose by energy range is possible based on the type of facility and predominant radionuclides such as intermediate (>100 keV) energies for all facilities handling activation and fission product nuclides, primarily lower (<100 keV) energy photons for plutonium facilities and lower energy photons for uranium fuel fabrication facilities. The recorded dose from the dosimeter response does not typically provide sufficient information to estimate discrete energy ranges. It is possible to examine the energy response characteristics of the respective multi-element dosimeters but this analysis does not recognize the substantial uncertainties present in the workplace associated with shielding, radiation scattering and mixed radiation fields.

E.4.1.6 Missed Neutron Dose Prior to 1971

Prior to 1971, the NTA film used at SRS did not provide a claimant-favorable estimate of the neutron dose. Essentially all SRS workers who may have received significant neutron dose did receive a measured photon dose. As such, using a ratio of the potential neutron dose to the measured photon dose is done as a claimant-favorable option to reconstruct an individual worker neutron dose. Table E-08 provides a summary of the recommended neutron to photon ratios for the different process areas. As can be determined from E-09, the recommended method to apply the ratio is as a log normal distribution using the geometric mean and geometric standard deviation. Caution, however, should be employed when using Table E-09 and consideration should be given to an individual energy employee's neutron to photon ratio if there is sufficient individual data. For example if an individual has recorded neutron dose either prior to or after the estimation period that has a higher neutron to photon ratio than what is provided in the table, then the higher neutron to photon ratio should be used provided that the work assignments were similar for the two time periods. The values in Table E-09 should be used when there is known technical limitations in the neutron dosimetry technology or no neutron monitoring data.

Table E-09 Neutron to Photon Ratios by SRS Process

Process	Description / Buildings	Operations		Neutron to Photon Ratio		
		Begin	End	Geometric Mean (GM)	Geometric Standard Deviation (GSD)	Upper 95th%
Fuel Fabrication	Produced reactor fuel and target assemblies. Intermittent production mixed oxide fuel and plutonium – aluminum alloy targets.					
	(321-M)			(a)	(a)	(a)
Reactors	During Reactor Operation: Low-level neutron exposure through shielding. In addition, some intermittent neutron exposure from mixed oxide fuel and targets.					
	(105C)	3/1955	1/1987	0.18	2.52	0.82
	(105K)	10/1954	1/1988			
		1992	7/1992			
	(105L)	7/1954	2/1968			
		1985	6/1988			
	(105P)	2/1954	8/1988			
	(105R)	12/1953	6/1964			
Plutonium Production	Plutonium Finishing Process: Plutonium enters the process as PuF_4 and is then fired into production pucks. Work is primarily conducted in glove boxes with predominant close anterior exposure to workers. Radiation levels at the beginning of the process are fairly constant while levels at end of process are closely related to production levels.					
	Plutonium Finishing Process (221-H, B-Line)	11/1954	1988	0.91	2.84	5.05
	Plutonium Finishing Process (221-F, B-Line)	10/1955	1988	0.36	2.52	1.65
	Plutonium Storage					
	Production Control Laboratory (772-F)	1955				
	Plutonium Fuel Facility (235-F)	1955				
Radionuclide Production and Calibration	Radiation characteristics are highly dependent upon the radionuclide being produced. SRS has produced several radionuclides including ^{238}Pu, ^{239}Pu, americium, curium, californium, etc. At the calibration facility radiological instruments are calibrated using a variety of sources with different neutron energy spectra.					
	Californium Production Facility (773-A)			0.62	2.29	2.41
	Calibration Facility (736-A)			(b)	(b)	(b)

(a) During Pu-Al target production, workers with potential for neutron exposure wore NTA film, thus LOD values listed in table 5.5.2-1 should be used.

(b) Neutron to photon ratio for the calibration facility is not expected to be greater than the ratio observed in the californium production facility. As a claimant favorable approximation the Californium Production facility ratio can be used.

Due to potential for photon missed dose effecting the neutron to photon ratio, when applying the ratio in Table E-09, the photon dose should be the combination of both the measured photon dose (Dosimeter Dose) and the photon missed dose. When overestimating dose reconstructions, the upper 95% ratio can be used as a claimant-favorable worst case ratio and the associated neutron uncertainty can be omitted. In underestimating dose reconstructions, the geometric mean can be used as a best case ratio.

E.4.1.7 Missed Neutron Dose After 1970

Following SRS implementation of the TLND on January 1, 1971 recorded neutron dose has been reasonably accurate. As such, OCA-IG-001 guidance on missed dose should be followed in accordance with the LOD values presented in Table E-10.

Table E-10 SRS TLND Limits of Detection

Dosimeter Type	Dates of Use		Exchange Frequency	Reported LOD (mrem)	Estimated LOD (mrem)	Max. Annual Missed Dose (mrem)
	Begin	End				
SRS Hoy TLND	1/1/1971	12/31/1994	Monthly (n=12)	10[(c)]	20	240
Panasonic TLND	1/1/1995	Present	Monthly (n=12)	15	15	180

(a) Based on Wilson et al (1990)
(b) Correction for estimated 25% under response of NTA film for some facilities
(c) Minimum Recorded Dose not necessarily LOD

E.4.1.8 Organ Dose Equivalent

Once the adjusted photon and neutron doses have been calculated for each year, the values are used to calculate the organ dose distribution for the primary organ of interest identified in the claim. Table E-05 summarizes default workplace geometries. These can be used in case more applicable values (NIOSH 2002) cannot be determined. A range of reasonable estimates may be evaluated to arrive at a claimant-favorable selection.

Table E-11 Default Exposure Geometries to Calculate Organ Dose

Claim Status[(a)]	Job Category [(b)]	Exposure Geometry	Percentage [(c)]
Non-compensable	All	AP	100%
Compensable - Workers	All	AP	50%
		ROT	50%
Compensable - Supervisors	All	AP	50%
		ISO	50%

(a) Specific time spans for the various SRS facility operation can be found in the section 5.
(b) More than one job category may be needed for longer-term employed workers.
(c) Apply this percentage to the dose conversion factor (NIOSH 2002a, Appendix B) to arrive at the total organ dose equivalent from the adjusted recorded dose.

E.4.2 Parameter #2

Parameter #2 is the standard deviation of the normal distribution for the organ dose. The individual dose result for each dosimeter exchange period will be available to calculate the mean and standard deviation for each year. If not available, the adjusted organ dose can be used for each year and a default standard deviation value used for parameter #2.

E.4.3 Organ Dose Conversion Factors

OCAS-IG-001 Appendix A (NIOSH 2002) contains a detailed discussion of the conversion of measured doses to organ dose equivalent, and Appendix B contains the appropriate dose conversion factors (DCFs) for each organ, radiation type, and energy range based on the type of monitoring performed. In some cases, simplifying assumptions are appropriate.