



## ORAU TEAM Dose Reconstruction Project for NIOSH

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<p>Document Title: <b>Site Profile for Sandia National Laboratories in Albuquerque, New Mexico, and the Tonopah Test Range, Nevada</b></p>	<p>Document Number: ORAUT-TKBS-0037  Revision: 00  Effective Date: 06/22/2007  Type of Document: TBD  Supersedes: None</p>
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New   
 Total Rewrite   
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**PUBLICATION RECORD**

<b>EFFECTIVE DATE</b>	<b>REVISION NUMBER</b>	<b>DESCRIPTION</b>
06/22/2007	00	New approved site profile for the Sandia National Laboratories in Albuquerque, New Mexico and the Tonopah Test Range, Nevada. Incorporates formal internal and NIOSH review comments. There is a reduction in assigned dose and no PER is required. Training required: As determined by the Task Manager. Initiated by Jack E. Buddenbaum.

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**ACRONYMS AND ABBREVIATIONS**

ACPR	Annular Core Pulse Reactor (later re-named ACRR), TA-V
ACRR	Annular Core Research Reactor, TA-V
AEC	U.S. Atomic Energy Commission
AMAD	activity median aerodynamic diameter
AP	anterior-posterior
AT&T	American Telephone and Telegraph
Bq	becquerel (1 disintegration per second)
BZA	breathing-zone air
CAM	continuous air monitor
CEP	Controls for Environmental Pollution
CFR	Code of Federal Regulations
cGy	centigray
CHP	Certified Health Physicist
Ci	curie
cm	centimeter
cpm	counts per minute
CWL	Chemical Waste Landfill, TA-III
d	day
DAC	derived air concentration
DCF	dose conversion factor
DOE	U.S. Department of Energy
dpm	disintegrations per minute
DU	depleted uranium (less <sup>235</sup> U than NU, nominally 0.72%)
ECF	Explosive Component Facility
EEOICPA	Energy Employees Occupational Illness Compensation Program Act of 2000
EPA	U.S. Environmental Protection Agency
ESE	entrance skin exposure
GIF	Gamma Irradiation Facility
GSD	geometric standard deviation
Gy	gray
HCF	Hot Cell Facility
HERMES II	high-energy, pulsed, field-emission, electron-beam or bremsstrahlung gamma ray generator (1968)
HPGe	hyperpure germanium
hr	hour
HT	elemental tritium (tritiated gas)
HTO	tritium oxide (water or water vapor)
HVL	half-value layer
HydraMITE II	a dual-line, high-energy, short-pulse source for electron-beam or bremsstrahlung X-ray (1984)
ICP-MS	inductively coupled plasma mass spectroscopy
ICRP	International Commission on Radiological Protection
IMBA	Integrated Modules for Bioassay Analysis

in.	inch
IREP	Interactive RadioEpidemiological Program
kCi	kilocurie
keV	kiloelectron-volt, 1,000 electron-volts
KPA	kinetic phosphorescence analysis
kVp	peak kilovoltage, applied kilovoltage
kW	kilowatt
L	liter
LANL	Los Alamos National Laboratory
LASL	Los Alamos Scientific Laboratory
LAT	lateral
L <sub>c</sub>	decision level
LLNL	Lawrence Livermore National Laboratory
LLW	low-level waste
LSC	liquid scintillation counter
m	meter
mA	milliampere
MABE	Megamp Accelerator and Beam Experiment (1986; also known as Gemini)
mAs	milliampere-second
MBq	megabecquerel
mCi	millicurie
MDA	minimum detectable activity
MDL	minimum detection limit
MDTA	minimum detectable true activity
MeV	megaelectron-volt, 1 million electron-volts
mg	milligram
mGy	milligray
min	minute
ml	milliliter
mm	millimeter
MOU	memorandum of understanding
MPC	maximum permissible concentration
mR	milliroentgen
mrad	millirad
mrem	millirem
MSMA	maximum significant measured activity
MT	metal tritide (tritium bound to metallic compounds, such as hafnium)
MW	megawatt
MWL	Mixed Waste Landfill, TA-III
n	neutron
nCi	nanocurie
NCRP	National Council on Radiation Protection and Measurements
NDA	no detectable activity
NIOSH	National Institute for Occupational Safety and Health
NTA	nuclear track emulsion, type A
NTS	Nevada Test Site
NTTR	Nevada Test and Training Range
NU	natural uranium

OBT	organically bound tritium
ORAU	Oak Ridge Associated Universities
PA	posterior-anterior
PBFA	Particle Beam Fusion Accelerator
pCi	picocurie
PFG	photofluorography
PHA	pulse height analysis
POC	probability of causation
R	roentgen
RAS	radiometric alpha spectroscopy
RMS	root mean square
RMWMF	Radioactive and Mixed Waste Management Facility, TA-III
RPID	Radiation Protection Internal Dosimetry
RWP	radiological work permit
s	second
SABRE	Sandia Accelerator and Beam Research Experiment
SER	Sandia Engineering Reactor (TA-V, circa 1957, highly enriched uranium fuel)
SERF	Sandia Engineering Reactor Facility
SID	source-to-image distance
SNL	Sandia National Laboratories
SNL-NM	SNL Albuquerque, New Mexico, facilities
SNL-NV	SNL TTR facilities
SPEED	high-energy, very short pulse bremsstrahlung X-ray accelerator (1983)
SPHINX	Short Pulse High Intensity Nanosecond X-Radiator
SPR	Sandia Pulse Reactor (Facility), TA-V
SRDB Ref ID	Site Research Database Reference Identification (number)
SSD	source-to-skin distance
Sv	sievert
TA	Technical Area
TBD	technical basis document
TLD	thermoluminescent dosimeter
TRIGA	Training, Research, Isotopes General Atomics
TTR	Tonopah Test Range
U.S.C.	United States Code
wt %	weight percent
yr	year
γ	gamma particle or ray
μCi	microcurie
μg	microgram
μm	micrometer
§	section or sections

## 1.0 INTRODUCTION

Technical basis documents and site profile documents are not official determinations made by the National Institute for Occupational Safety and Health (NIOSH) but are rather general working documents that provide historic background information and guidance to assist in the preparation of dose reconstructions for particular sites or categories of sites. They will be revised in the event additional relevant information is obtained about the affected site(s). These documents may be used to assist NIOSH staff in the completion of the individual work required for each dose reconstruction.

In this document the word “facility” is used as a general term for an area, building, or group of buildings that served a specific purpose at a site. It does not necessarily connote an “atomic weapons employer facility” or a “Department of Energy [DOE] facility” as defined in the Energy Employees Occupational Illness Compensation Program Act [EEOICPA; 42 U.S.C. § 7384l(5) and (12)]. EEOICPA defines a DOE facility as “any building, structure, or premise, including the grounds upon which such building, structure, or premise is located ... in which operations are, or have been, conducted by, or on behalf of, the Department of Energy (except for buildings, structures, premises, grounds, or operations ... pertaining to the Naval Nuclear Propulsion Program)” [42 U.S.C. § 7384l(12)]. Accordingly, except for the exclusion for the Naval Nuclear Propulsion Program noted above, any facility that performs or performed DOE operations of any nature whatsoever is a DOE facility encompassed by EEOICPA.

For employees of DOE or its contractors with cancer, the DOE facility definition only determines eligibility for a dose reconstruction, which is a prerequisite to a compensation decision (except for members of the Special Exposure Cohort). The compensation decision for cancer claimants is based on a section of the statute entitled “Exposure in the Performance of Duty.” That provision [42 U.S.C. § 7384n(b)] says that an individual with cancer “shall be determined to have sustained that cancer in the performance of duty for purposes of the compensation program if, and only if, the cancer ... was at least as likely as not related to employment at the facility [where the employee worked], as determined in accordance with the POC [probability of causation<sup>1</sup>] guidelines established under subsection (c) ...” [42 U.S.C. § 7384n(b)]. Neither the statute nor the probability of causation guidelines (nor the dose reconstruction regulation) define “performance of duty” for DOE employees with a covered cancer or restrict the “duty” to nuclear weapons work.

As noted above, the statute includes a definition of a DOE facility that excludes “buildings, structures, premises, grounds, or operations covered by Executive Order No. 12344, dated February 1, 1982 (42 U.S.C. 7158 note), pertaining to the Naval Nuclear Propulsion Program” [42 U.S.C. § 7384l(12)]. While this definition contains an exclusion with respect to the Naval Nuclear Propulsion Program, the section of EEOICPA that deals with the compensation decision for covered employees with cancer [i.e., 42 U.S.C. § 7384n(b), entitled “Exposure in the Performance of Duty”] does not contain such an exclusion. Therefore, the statute requires NIOSH to include all occupationally derived radiation exposures at covered facilities in its dose reconstructions for employees at DOE facilities, including radiation exposures related to the Naval Nuclear Propulsion Program. As a result, all internal and external dosimetry monitoring results are considered valid for use in dose reconstruction. No efforts are made to determine the eligibility of any fraction of total measured exposure for inclusion in dose reconstruction. NIOSH, however, does not consider the following exposures to be occupationally derived:

- Radiation from naturally occurring radon present in conventional structures
- Radiation from diagnostic X-rays received in the treatment of work-related injuries

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<sup>1</sup> The U.S. Department of Labor is ultimately responsible under the EEOICPA for determining the POC.

## 1.1 PURPOSE

This site profile for Sandia National Laboratories (SNL) provides background information in relation to dose reconstruction for SNL workers at the Albuquerque, New Mexico (SNL-NM), site and the Tonopah, Nevada, (SNL-NV) site on the Tonopah Test Range (TTR). The site profile also provides details regarding past and current SNL practices used to assess radiation exposures and environmental radiation levels at the facilities. In this document, SNL refers either to the Laboratories as a whole or to the Albuquerque site; SNL-NV refers to the TTR site.

## 1.2 SCOPE

Section 2.0 provides an overview of historical operations at SNL-NM and SNL-NV that have involved potential external or internal radiation exposures to workers.

Section 3.0 provides information about the doses that individual workers could have received from X-rays that were required as a condition of employment. These X-rays included preemployment and routine chest X-rays during their required physical exams.

Section 4.0 presents environmental dose information for workers who have received doses when working outside the buildings on the site from inhalation of radioactive materials in the air, direct radiation from plumes (immersion dose from radioactivity in the air), contact with radioactive particles on the skin, and from direct exposure to radionuclides that were incorporated in the soil.

Section 5.0 discusses the internal dosimetry program at SNL, including discussions of *in vitro* minimum detectable activities (MDAs). Details of the monitoring techniques and programs are also presented in this section. Information on the specific radionuclides that workers could have been exposed to in each of the various facilities is presented in this section as well.

Section 6.0 discusses the program for measuring skin and whole-body doses to the workers. The methods for evaluating external doses to workers have evolved over the years as new techniques and equipment have been developed.

Attributions and annotations, indicated by bracketed callouts and used to identify the source, justification, or clarification of the associated information, are presented in Section 6.0.

## 2.0 SITE DESCRIPTION

### 2.1 PURPOSE AND SCOPE

This section provides a site description that contains technical basis information for the evaluation of the total occupational dose for EEOICPA claimants. It describes SNL facilities and processes and it details the historical information related to worker internal and external exposures.

### 2.2 INTRODUCTION

SNL had its origin as a satellite support site for the Los Alamos Scientific Laboratory (LASL; later renamed Los Alamos National Laboratory [LANL]). The LANL Z-Division was established in July 1945 to handle weapons development, testing, and bomb assembly for the Manhattan Engineer District (Ullrich 1998). In late fall of 1945, some units of Z-Division were moved to the current SNL-NM site near Albuquerque. In 1948, Z-Division became a separate branch of LANL and a year later it was renamed Sandia Corporation. Its functions included engineering research, development, and mechanical production of the specialized and strategic systems components needed to support U.S.

national security. The site became known as Sandia Laboratories in 1971 and then underwent another name change in 1979 to SNL [1].

## 2.3 SITE ACTIVITIES AND PROCESSES

From 1948 to 1959, three national laboratories were coordinated in nuclear weapons design and construction activities: (1) LANL, (2) SNL, and (3) Lawrence Livermore National Laboratory (LLNL). The three laboratories were part of the larger network of U.S. Atomic Energy Commission (AEC)-controlled sites that completed the manufacture, testing, and storage of nuclear weapons for the national stockpile. As described by Leland Johnson (SNL 1997):

*"Los Alamos and Lawrence Livermore design the high explosive and nuclear system package, while Sandia designs the rest of the nuclear bomb or warhead, including the arming, fusing, and firing systems along with other essential components. In essence, Sandia 'weaponizes' the nuclear systems designed at its partner laboratories."*

Activities at SNL during the 1950s did not involve significant use of radioactive materials. Most components were handled away from the SNL-NM site in this early era. Monitoring records for personnel that worked on- or off-site are maintained at SNL. Activities included extensive field testing of weapons components until 1958, which began again in 1961. The first nuclear reactor at SNL, the Sandia Engineering Reactor Facility (referred to as SER or SERF in historical records), went on line in 1957 and was used for material testing.

In the early 1960s, several high-energy accelerators were added at SNL to facilitate research on the effects of pulsed radiation on a wide variety of materials. The interest in the effects of pulses of higher intensity radiation (higher than could be sustained continuously with either the SERF reactor or the accelerators of the era) led to the additional installation and startup of the Sandia Pulse Reactor (SPR-I) in 1961. This unit was a GODIVA-type reactor design that allowed running the core up to relatively high energy levels for very short periods to investigate material damage under controlled conditions (SNL 1983). Over the succeeding decades, this design was upgraded to increasingly higher energy-level capacities to determine when exposed target materials fail or decompose. Information gained from these tests was used to improve design specifications for materials and systems exposed to various radiation environments.

Over the years, the laboratory's mission expanded and SNL became important in areas beyond nuclear weapons development. New activities included working on technologies to monitor nuclear testing after the treaty of 1963, working with the National Aeronautics and Space Administration to enhance the safety of aerospace nuclear power systems, and developing conventional weapons and intrusion sensors for use in the Vietnam War. National and international events, including the energy crisis and the terrorist acts at the Munich Olympics of the early 1970s, caused SNL to become involved in new areas of energy research and in physical security and safeguards for facilities. However, SNL continued to have responsibilities in developing new nuclear and other weapons as well as maintaining the safety and reliability of the existing nuclear stockpile (SNL 1997).

Although international arms control efforts increased throughout the 1970s and 1980s, concerns remained about the impact of pulses or bursts of high radiation on both materials and functionality of electromagnetic and electronic systems. The relatively small energy level flash X-ray and heavy ion accelerators of the early 1970s were being rapidly augmented by 1- and 2-MeV machines. The continuing demand in the 1980s and 1990s for larger capacity and higher energy accelerators for testing and stressing electronic components with exposure to pulsed electron and ion accelerators

and flash X-ray systems led to continual development changes and additions of higher energy accelerators at SNL during that period.

Table 2-1 provides a summary chronology of SNL programs and events.

Table 2-1. Chronology of significant SNL programs and events.

Year	Activity
1945	Z-Division created at LANL to assemble stockpile materials.
1946	Los Alamos Z-Division moved to current SNL location in Albuquerque. Building 828 constructed in TA-1 for testing of weapons components.
1948	TA-2, high explosive final assembly area for nonnuclear components constructed; assembly continued for many years, primarily in Buildings 901, 904, and 907.
1949	Name change to Sandia National Laboratory, separate from LANL.
1953	Large centrifuge, rocket sled track, vibration facility completed at TA-3.
1959	Mixed Waste Landfill (MWL) (1959 to 1988) at TA-3 for disposal of low-level mixed waste. Contained uranium (depleted, natural, and enriched), thorium, tritium beds, liquid scintillation cocktails, plutonium wastes.
1957	SNL given delivery responsibility for all training units (with high explosive), inert training units (no high explosive), and inert samples of weapons in early design stages until 1959. Assembly moved out of Area II.
1960	TA-2 converted into an explosive devices research and development area.
1960–1980s	TTR (NV) used to test nonnuclear systems and components and underground nuclear experiments.
1960–1980s	Kauai Test Facility, HI used for nonnuclear weapons testing. No radionuclides used.
1964	Component test facility for neutron generators added to TA-2 (Building 935). Potential radiation exposures included neutrons and tritium.
1967	Building 919 added to TA-2 for neutron generator (containing tritium) testing.
1980s	Operations commenced in TA-4 with pulsed power accelerators.
1985	Gamma Irradiation Facility (GIF) at TA-5; north and south cells. Sources (1) Co-60 49 kCi (1985) and (2) Cs-137 163 kCi (1985) in north cell, and (1) 150 kCi Co-60 (upgraded to 300 kCi in 1985) in south cell.
1996	ACRR and SPR reactor facilities online in TA-5. ACRR fuel is U enriched to 35% with 21.5 wt % UO <sub>2</sub> and 78.5 wt % BeO. SPR II and III fuel is U enriched to 93%.
1996	HCF established to handle and examine radioactive materials (Co-60, Cs-137 and miscellaneous fission products) from the SNL reactors and experiments.

## 2.4 MAJOR FACILITIES AND ACTIVITIES

Early on, it became apparent that SNL would need to grow to house the increasing staff necessary to meet scientific demands. SNL currently consists of five technical areas (TAs) and several test areas. Each TA has its own distinctive operations, but the operations of some groups at SNL can span more than one TA. A description of each area is given below. Table 2-2 illustrates each area and the notable facilities and buildings along with major radionuclides and years of operation. Table 2-3 lists the specific areas of radionuclide use or radiation generation and the activity or power level.

### 2.4.1 TA-1

TA-1 operations have been dedicated primarily to three activities: (1) the design, research, and development of weapon systems; (2) limited production of weapon system components; and (3) energy research programs. TA-1 facilities include a Cockcroft-Walton electron accelerator and a Van de Graaff generator as well as heavy ion/proton accelerators, a fexitron (an early flash X-ray machine), and another higher energy X-ray system with bremsstrahlung beam to irradiate electronic

components. These systems often served as pilot versions for much larger machines constructed in more remote areas (TA-4 and TA-5).

Table 2-2. Area information and parameters.<sup>a</sup>

Area (bldg)	Description of process	Years of operation	<u>Radionuclides</u> M = major, L = likely (AMAD=default 5 um unless given <sup>b</sup> )	<u>Radiation types</u> B = beam types
<b>TA-1 Electron/ion beam accelerators</b>				
B-884	Cockroft-Walton(s) (electron accelerator)	Late 1950s– present (2006)	L = Target dependent (e.g. H-3)	B = electron, X-ray L = target dependent (e.g. H-3, DU, Be)
B-803	Van de Graaff(s) (electron accelerator)	1958–present (2006)	L = Target dependent (e.g. H-3)	B = electron, X-ray (400 keV-2 MeV)
B-803 B-672	Heavy Ion/Proton Accelerators ~ (100 kV)	1968+	NA	B = protons, ions and X-rays
B-672	Fexitron	1970–1973+	NA	B = flash X-ray
B-642	NERUES	1971+	NA	B = X-ray, gamma, bremsstrahlung
<b>TA-1 Manufacturing facilities</b>				
B-802, B-805, B-870, B-884	Kaman Neutron Generators Mfg. (NGMF)	- Present?	Po-Be, Am-Be, Pu-Be	(neutrons)
B-800s	Chemistry Labs	1949-present	H-3	NA
B-805	LICA –Low Irradiation Calibrator Apparatus	>2003–2005	M = Co-60, Cs-137	Gamma, beta
B-818, B-828	Radiation Standards Calibration Facility	1998–current	M = Co-60, Cs-137	Gamma, beta
B-869	Toxic Metals Machine Shop	~1959–1994	M = DU, Pu	NA
B-883	6 MeV Ion generator	1984–1996		B = ion beam, X-rays
B-858	Sealed sources in micro-electronics development	(Single incident)	L= wide variety, but sealed.	NA
<b>TA-1 Medical services</b>				
B-831, Rm-130	Medical X-ray (Picker GX-325-PX350 tube) 300 mA @ 125 kVp	1949–present, (registered in 1978)	NA	X-ray
<b>TA-2 Repair/test services</b>				
B-935	Neutron Generator	1969–1993	L= only if damaged: Am-241, Po-210, Pu -239	(neutrons)
<b>TA-3 Waste destruction, disposal, transfer</b>				
	“Leaking Cask”	{Single incident}	Sr-90, Cs-137	(See SNL 1983)
B-6583	Classified Destruction Facility	1957–1988	Transuranic elements, others	
B-6920	Radiological and Mixed Waste facility	1949–1986	Cs-137, DU	(See SNL 1983)
<b>TA-4 Electron/ion/X-ray beam accelerators</b>				
B-981–B-983	Particle beam fusion accelerator(PBFA)-I	1987–1995	N-13, O-15 gases	B = electrons, gamma rays; target/ neutrons
B-981	SATURN (succeeded PBFA-I, preceded PBFA-II)	1967–1996+	N-13, O-15 gases	B = electrons, gamma rays; target/neutrons
B-983	PBFA-II (<30 MeV)	1993–present	N-13, O-15 gases (<0.042 Ci/yr, N-13, 0.005 Ci/yr O-15)	B = electrons, gamma rays; target/neutrons
B-983	Z-machine (modified from PBFA-II)	1996–present	N-13, O-15 gases	B = gamma rays, target/neutrons
B-961	MITE pulsed X-ray (6 MeV) [4 units]	Present	NA	B = X-ray, gamma rays
B-970	HERMES III (higher energy version of HERMES II) (< 20 MeV)	~1988–1998	N-13, O-15 gases (<2.3 & 0.03 Ci/yr)	B = gamma rays, target/neutrons
B-970	Sandia Accelerator and Beam Research Experiment (SABRE; 6-12 MeV)	1998–present	N-13, O-15 gases (<0.0058 Ci/yr, N-13.)	B = gamma rays, target/neutrons
B-981	SPEED (1.0 MeV)	1983–1986	N-13, O-15 gases	B = gamma rays, target/neutrons
B-981	SPHINX	1992–present	N-13, O-15 gases	B = gamma rays, target/neutrons
B-961	TESLA	1998–current	NA	B = X-rays
<b>TA-4 Repair/test facility</b>				
B-905	Neutron Generator =Test Equipment	1959–1997+	L = only if generators, Damaged: Am-241, Po-219, Pu-239	B= neutrons

Area (bldg)	Description of process	Years of operation	Radionuclides M = major, L = likely (AMAD=default 5 um unless given <sup>b</sup> )	Radiation types B = beam types
<b>TA-5 Electron/ion beam accelerators</b>				
B-6581	Febetrans (Flash X-ray system-< 2 MeV)	1967+		B = Flash X-ray (~2 MeV)
B-6580	Proto - I (1st generation high powered short pulse accelerator)	1972--1976	Target dependent (e.g., H-3)	B = electrons, X-rays
B-6580	Proto - II (2nd generation high powered short pulse accelerator)	1976--1998	Target dependent (e.g., H-3)	B = electrons, X-rays
B-6580	Pelletron Facility --variable energy high stability dc electron beam generator (1 MeV)	1968--1991	Target dependent (e.g., H-3)	B = electrons, X-rays
B-6580	HERMES I, II (field emission electron beam or bremsstrahlung X-ray accelerator)	1968--1988	Target dependent (e.g., H-3)	B = electrons, ions, bremsstrahlung X-ray
B-6580	Relativistic Electron Beam Accelerator (REBA) = Z-machine (3.2 MV)	1994--present	Target dependent (e.g., H-3)	B = electrons, Bremsstrahlung X-ray
B-6580	Hydra-HydraMITE I & II-- dual transmission line for high energy short-pulse electron, bremsstrahlung X-ray unit (1 MV)	1977, 1984+	Target dependent (e.g., H-3)	B = electrons, ions, bremsstrahlung X-ray
B-6580	REHYD = Heavy Ion Accelerator variable energy, positive ion combination of REBA and HydraMITE (1.3 MV)	1988--1998+	Target dependent (e.g., H-3)	B = electrons, ions, bremsstrahlung X-ray
<b>TA-5 Reactors</b>				
B-6581	SER (5 MW) Facility (~NPR@INL)	1958--1979	M= Ar-41; L = All reactor-produced nuclides	B = prompt gamma, beta
B-6588	ACRR (600 kW) also operated as Pulsed Reactor (ACPR) (<15,000 MW) [TRIGA-type]	1968--1998	M= Ar-41 L = All reactor-produced nuclides	B = prompt gamma, beta
B-6588	ACRR = Mo-99 Production (600 kW)	1998--present	M= Ar-41, Tc-Mo-99 L = All reactor-produced nuclides	B = prompt gamma, beta
B-6590	SPR-I [GODIVA-type]	1961--1975	M= Ar-41 L = All reactor-produced nuclides	B = prompt gamma, beta
B-6590	SPR-II [GODIVA-type] (<130,000 MW pk.)	>1962--present	M= Ar-41 L = All reactor-produced nuclides	B = prompt gamma, beta
B-6596	SPR-III (KIVA=SPR-III) [GODIVA-type] (< 170,000 MW pk.)	2003--present	M= Ar-41 L = All reactor-produced nuclides	B = prompt gamma, beta
<b>TA-5 Irradiation and calibration support laboratories</b>				
B-6580, B-6581	HCF	1988--present	Target dependent: (e.g., metals from reactor, e.g., Cs-137, Ce-144, and H-3)	NA
B-6591	GIF (~ 900 Ci Co-60 and similar Cs-137 high activity sources for irradiation experiments)	1962--1998	M =Co-60 and Cs-137	NA
B-6591, B-6598	NEW GIF--combined with LICA--more flexible sources--(Co-60 and Cs-137 high activity sources for irradiation experiments)	1998?--present	M =Co-60 and Cs-137	NA
<b>Coyote Test Facility</b>	Thunder Range--source metals after explosion and fire tests of weapons, containers	1969--1994	M = DU	NA

a. NA = not available.

b. AMAD = activity median aerodynamic diameter.

Table 2-3. Magnitude of activity<sup>a</sup>.

Area/facility	Radionuclide/beam	Activity/power level
<b>TA-1</b>		
800 Series Buildings	Po-Be, Am-Be, Pu-Be, Co-60, DU	NA
Cockroft-Walton		100 keV +
Van de Graaff		400 keV -- 2 MeV
Heavy Ion Accelerators		100 kV
Fexitron	X-ray	NA
NERUES	X-ray, gamma, bremsstrahlung	NA
<b>TA-2</b>		
900 Series Buildings	Explosives; nonnuclear	NA

Area/facility	Radionuclide/beam	Activity/power level
<b>TA-3</b>		
B-6920 (waste facility)		NA
<b>TA-4</b>		
EBFA/PBFA I	Prompt gamma, O-15, N-13	NA
PBFA II	Prompt gamma, O-15, N-13	Maximum of 30 MeV with 5 shots/wk; 0.000168 Ci N-13 and 0.00002 Ci O-15 per shot (<0.042 Ci/yr N-13, 0.005 Ci/yr O-15).
RHEPP I	X-rays	
HERMES III	Prompt gamma, O-15, N-13	2.32 Ci/yr of N-13 and 0,030 Ci/yr for O-15 based on stack emissions / 20 MeV endpoint voltage
SABRE	Prompt gamma, N-13	1.16E-05 Ci Ni-13 per shot / 12 MeV maximum
STF	Prompt gamma, O-15, N-13	NA
ALIAS	Prompt gamma, O-15, N-13	NA
TROLL	Prompt gamma, O-15, N-13	NA
PI-112	Prompt gamma, O-15, N-13	NA
SATURN	Prompt gamma, O-15, N-13, H-3	4.75 µCi H-3 between 10/29 and 11/09, 1990. No H-3 between 1990 and 1996.
SPHINX	Prompt gamma	NA
PROTO II (after late 1980s)		NA
MITE	X-ray, gamma	6 MeV
<b>TA-5</b>		
ACRR/ACPR	Ar-41, Tc-99, Mo-99, reactor-produced nuclides	600 kW (ACRR); <15,000 MW (ACPR)
SPR I, II, III	Ar-41, reactor-produced nuclides	<130,000 MW (SPR I) <170,000 MW (SPR II)
HCF	Co-60, Cs-137, mixed fission products	NA
GIF	Co-60, Cs-137	NA
HERMES I, II	Electrons, bremsstrahlung	NA
REBA	Electrons, bremsstrahlung	3.2 MV
Febetrans	Flash X-ray	< 2 MeV
PROTO I, II	Electrons, X-rays	NA
Pelletron	Electrons, X-rays	1 MeV
Hydra-HydraMITE	Electrons, bremsstrahlung	1 MV
REHYD	Electrons, bremsstrahlung	1.3 MV
B-6591 (GIF)	Co-60, Cs-137	900 Ci Co-60, Ci Cs-137
<b>San Diego Test Facility</b>		
Early underwater missile tests (nonradioactive dispersal)	NA	NA
<b>TTF</b>		
Missile testing	DU contamination	NA
<b>NTS</b>		
Weapons testing	Fission Products (e.g., Sr/Y-90, Cs-137)	NA
<b>Kauai, HI, Test Facility</b>		
No-radioactive materials, missile tests	NA	NA
<b>Livermore, CA</b>		
Tritium - Experimental Laboratory	H-3	NA

Area/facility	Radionuclide/beam	Activity/power level
<b>Vandenberg, CA</b>		
Nose cone ablation testing	Ta and Se	NA

a. NA = Not Available or Applicable

#### 2.4.1.1 Buildings with Radionuclide Activity

Because TA-1 was the site area initially developed, it contains a large number of buildings of which at least 75 have been routinely included in contamination survey programs over the last 20 yr. Many of the buildings have laboratories that incidentally use small radionuclide quantities or that have ion beam or X-ray machines used to support electronic and mechanical component manufacture. A few locations (B-802, B-805, B-870, and B-884) have been devoted to neutron source manufacturing over a very long period.

Building 803 housed a Van de Graaff generator and Building 884 contained a Cockroft-Walton electron accelerator beginning in the late 1950s to the present. These generators were used in early experiments designed to expose materials to large exposures of gamma and electron beam radiation during controlled experiments. Buildings 818 and 828 housed the primary radiation calibration facilities used for primary calibration of all dosimetry meters used on the SNL site over the last 50 yr. These several uses have persisted until the present in TA-1. However, as noted above, the need for larger radiation sources such as research reactors and large accelerators and hot cell facilities to support them for a much larger scale of radiation exposure experiments required the development of facilities in TA-4 and TA-5.

#### 2.4.2 TA-2

TA-2 is a 45-acre (180,000 m<sup>2</sup>) facility south of TA-1, established in 1948 for the assembly of chemical high-explosive main charges for nuclear weapons and later for production-scale assembly of nuclear weapons (Ullrich 1998). Assembly activities continued from 1952 to 1957 when the work shifted to other sites. Test devices and weapon prototypes continued to be assembled in TA-2 for many years. In 1960, TA-2 was converted into an explosive devices research and development area, which continued until 1995 when all explosive devices were removed from the area. Buildings 901, 904, and 907 were the buildings associated with assembly operations.

##### 2.4.2.1 Buildings with Radionuclide Activity

In 1964, a component test facility was added as Building 935 to develop and test neutron generators. Building 919 was added in 1967 as an explosive device quality laboratory; tests on neutron generators continued to be conducted over the years. In general and based on review of the historical radiation protection program database, TA-2 has seen little use of radioactive materials.

#### 2.4.3 TA-3

Remote facilities were built in a new area, TA-3, 7 miles south of TA-2 for full-scale testing of weapons with and without explosives. TA-3 contains extensive design-test facilities such as rocket sled tracks, centrifuges, and a radiant heat facility (SNL 1998a). Other facilities in TA-3 include a paper destructor, the Melting and Solidification Laboratory, the radioactive and low-level waste landfills, and the MWL that operated from 1959 to 1988 (SNL 1996). Wastes disposed of in classified pits included uranium (depleted, natural, and enriched), thorium, barium, enriched lithium, tritium beds, neutron generator tubes and targets, plutonium contaminated wastes, and plutonium contaminated weapons test debris from the Nevada Test Site (NTS). Other than small buildings housing the operators and screening detectors to manage the waste disposal facilities, it does not

appear that any buildings in TA-3 had significant sources of radioactivity present for normal operations.

#### **2.4.3.1 Buildings with Radionuclide Activity**

Other than small buildings housing the operators and screening detectors to manage the waste disposal facilities in TA-3 (6583, 6920), it does not appear from the reviewed records that any of the buildings in this area had significant sources of radioactivity present for normal operations.

#### **2.4.4 TA-4**

TA-4 was opened to provide specialized remote research areas for pulsed-power and high-energy experiments (SNL 1998a). It consists of several inertial-confinement fusion research and pulsed-power research facilities, including the High Energy Radiation Megavolt Electron Source (HERMES-III), the Z Facility, the Short Pulsed High Intensity Nanosecond X-Radiator (SPHINX) Facility, and the Saturn Accelerator (Ullrich 1998).

##### **2.4.4.1 Buildings with Radionuclide Activity**

There are several principal buildings that house the largest and most powerful accelerators at the SNL site. These include B-961, B-970, B-981, and B-983. The specific machines located in each are identified in Table 2-2.

#### **2.4.5 TA-5**

TA-5 is a highly secure remote research area housing experimental and engineering nuclear reactors and several electron beam accelerators. It contains the Annular Core Research Reactor (ACRR) and the SPR (in two reactor facilities), an intense Gamma Irradiation Facility (GIF) (using  $^{60}\text{Co}$  and  $^{137}\text{Cs}$  sources), and the Hot Cell Facility (HCF) (DOE 1996). The ACRR was built to enable operation in both steady-state and pulsed modes; it was designated as the Annular Core Pulsed Reactor (ACPR) during the pulsed-mode operational periods (SNL 1983). After two decades of operation as a research reactor, the ACRR was transformed into a molybdenum-technetium radiopharmaceutical production unit for several years; it was idled in 2003.

##### **2.4.5.1 Buildings with Radionuclide Activity**

There are more than a dozen buildings in TA-5, which is primarily a location known for its research reactor and hot cell complex. Building numbers 6540, 6560, 6600-6630, and 9923-9940 are included in this complex. The new GIF with  $\sim 140\text{-kCi } ^{60}\text{Co}$  and  $\sim 200\text{-kCi } ^{137}\text{Cs}$  sources is in the Building 6591-6598 complex in TA-5 (SNL 1983). Other specific building locations for each of the reactors and the HCF are identified in Table 2-2.

#### **2.4.6 Other Areas**

In addition, SNL has test areas outside of the five TAs listed above. These test areas, including the Thunder Range in the canyons on the west side of the Manzanita Mountains, are known collectively as Coyote Test Field, which is southeast of TA-3.

The NTS was utilized in 1957 for testing of nuclear systems and bomb components in a limited series of aboveground tests (DOE 1997). After the Russians ended the initial agreement on a nuclear test ban in 1961, the U.S. reinitiated testing. All further testing occurred underground. A long series of underground nuclear tests were conducted between 1962 and 1973. Event-associated services were

provided by the Environmental Sciences Department of Reynolds Electrical & Engineering Company. Health physics support was supplied by SNL (SNL 1974a).

The TTR in Nevada was established in 1957 for testing of nonnuclear systems and bomb components. Many of these tests involved components containing depleted uranium (DU); there is some evidence of DU contamination at the site.

SNL conducted some nonnuclear weapons testing in Hawaii at the naval facility on Kauai. The facility was used to launch missiles carrying experimental, nonnuclear payloads (DOE 1997).

## **2.5 MAJOR SITE INCIDENTS**

A single incident of "serious overexposure" at SNL occurred in 1960. The event involved misuse of one of the new research Van de Graaff accelerators in Building 803. A single employee received a life-threatening exposure and serious hand injuries when the interlock protocol intended to prevent such situations was contravened. Additional monitoring equipment was installed and revised protocols were created; no similar events have occurred in the last 45 yr.

No other major site incidents were ever recorded at SNL.

## **2.6 HEALTH PROTECTION PRACTICES**

### **2.6.1 Personnel Monitoring**

#### **2.6.1.1 Badging**

From 1948 to 1958, the exposure monitoring program used simple film single-area film badges. Batches of these badges were periodically tested with a gridded table with pegs at specified distances from a central  $^{226}\text{Ra}$  source (traceable to the National Bureau of Standards). The data from readings of badges was manually recorded on 5- by 8-in. file cards. The format for recording data was eventually changed to punch cards so the data could be entered into a computerized database. In 1959, Eberline automated the readers and 4-spot film badges like those used at Oak Ridge were introduced. About the same time, Victoreen R Chambers were introduced into the calibration process to allow the precision of the calibration exposures to be more readily determined. Before 1948, dosimetry was provided by LANL. Discussions of film badges used after 1958 and before 1971 are presented in Section 6.0.

In May 1971, thermoluminescent dosimeters (TLDs) replaced all film badges. Before then, several radiation field-mapping studies for new source areas continued to use both film badges and TLDs to compare the familiar form of exposure data with the more sensitive TLDs.

#### **2.6.1.2 Area Monitoring**

In 1960, area monitors were in use in some areas, but had not yet been fully installed in the work area adjacent to the Van de Graaff generator in Building 803, where the serious overexposure incident occurred. After that incident, the system was widely expanded to serve as a warning system to supplement the badge program.

### **2.6.2 Radiological Exposure and Access Control**

SNL has changed over time from a laboratory where mostly classified activities took place to a relatively open campus where outside researchers from industry, academia, and other national

laboratories can gain access to the site's sophisticated facilities. In the early years, the majority of work was classified and access to facilities was strictly controlled by physical barriers.

### 3.0 **OCCUPATIONAL MEDICAL DOSE**

#### 3.1 **INTRODUCTION**

##### 3.1.1 **Purpose**

This section presents information that can be used by a dose reconstructor to estimate the dose received by a worker from occupational medical X-rays that were administered to the worker as part of their conditions of employment at Sandia National Laboratories (SNL).

##### 3.1.2 **Scope**

This section provides specific information concerning documentation of historical medical X-ray practices at SNL.

### 3.2 **EXAMINATION TYPES AND FREQUENCIES**

Detailed documentation of X-ray examination protocols is not available for SNL. Based on interviews with former X-ray technologists Kay Sanderville and Louise Bland and on review of employee medical records, a reasonably clear picture of the frequencies and types of radiographs taken at SNL over the years can be constructed (Stout 2005). Table 3-1 summarizes this information.

Table 3-1. Frequency and types of medical radiographs.

<b>Period</b>	<b>Type</b>	<b>Frequency</b>	<b>Comments</b>
1953–1966	Chest PFG and/or 14" x 17" PA chest	Preemployment/new hire + annual	Selected individuals could have received standard 14- by 17-in. PA chest radiographs in addition to PFG. Employee could choose to forego chest X-ray. Annual 14- by 17-in. chest X-ray required for respirator users. Any X-ray records in the claim file should be used in determining the type and frequency of X-ray procedures. In the absence of any records of X-ray procedures, the dose from PFG should be assigned on an annual basis for this period.
1967–1974	PA chest, 14 by 17 in.	Preemployment/new hire + biennial; some annual	Employee could choose to forego chest X-ray. Annual 14- by 17-in. chest X-ray for respirator users.
1975–present	PA chest, 14 by 17 in.	Preemployment/new hire + every 5 yr, some annual	Employee could choose to forego chest X-ray. Annual 14- by 17-in. chest X-ray for respirator users.
1953–1985	Lumbar spine	Preemployment/new hire	4 views: AP; LAT; AP angle at S1 level, LAT spot. Performed on small percentage of workers, probably according to job classification.

The table reflects the period assumed for some procedures to account for uncertainties in the specific period during which they could have occurred and thereby ensures favorability to claimants. Dose reconstructors should consider the frequencies listed in Table 3-1 to be default values favorable to the claimant for use unless other documented data in the claimant files are available.

From 1953 to 1985, SNL routinely performed preemployment or new hire chest X-ray examinations, but it appears that not all new hires received these chest X-rays because the individual could choose to forego this portion of the examination. Further, employees transferring to SNL from other

government agencies or contract facilities often brought their chest X-rays with them, so SNL did not repeat this procedure as a part of the new hire examination [2].

From the review of employee X-ray folders, the standard procedure through 1966 was apparently to use 4- by 5-in. chest photofluorography (PFG), with some employees receiving 14- by 17-in. posterior-anterior (PA) projections, perhaps in addition to the PFG. However, if an energy employee's record shows a distance of 72 inches then the exam was not a PFG and, therefore, a dose associated with a 14 x 17 exam should be assigned. Selected employees also received a series of four lumbar spine radiographs that consisted of an anterior-posterior (AP), a lateral (LAT), an AP angle at the S1 level, and an L5-S1 LAT spot film. The review of employee medical records indicated that SNL took PA chest films for most but not all employees on an annual or biennial frequency through 1966 [3]. In 1967, the frequency changed to biennial or triennial; to be favorable to the claimant, this analysis assumed a biennial frequency. In 1975, the chest X-ray frequency changed to a 5-yr cycle except for a relatively small number of employees in special surveillance programs and SNL discontinued lumbar spine radiographs for most job classifications. Employees with job classifications with higher risk for back injury such as maintenance personnel, heavy-equipment operators, and employees whose job duties required qualification to wear a respirator received lumbar spine or chest X-rays, respectively, on an annual basis.

By about 1983, the number of chest X-rays taken at SNL was reduced by about 50%; they were limited based on medical history and job classification. Routine lumbar spine X-rays were discontinued in the mid-1980s. To be favorable to the claimant, this analysis extended the date of this change through 1985 (as indicated in Table 3-1). In 1989, SNL again revised the physical examination protocol. Radiographs were limited to PA projections of the chest requested by a physician based on worker medical history and job classification [4].

### 3.3 TECHNICAL FACTORS THAT AFFECT X-RAY DOSE

A number of factors determine the dose from a medical X-ray procedure. For a standard medical radiographic unit with a tungsten target (anode) and focal spot of 1 to 2 mm, the major factors that affect the entrance skin *exposure*<sup>2</sup> (ESE) and organ dose include the basic machine settings used for the exposure, which include the applied kilovoltage of the beam (kVp), beam current (milliamperes), time of exposure (seconds), distance (centimeters), waveform, amount and kind of filtration used, and collimation or use of diaphragms. The following paragraphs discuss each of these factors; ORAUT (2005a) contains a more complete discussion. Other factors such as tube housing characteristics and leakage, type and speed of the X-ray film, development procedures, and screens and grids probably have little effect on organ doses and, thus, are not addressed below (see ORAUT 2005a).

Where X-ray exposure or dose measurements are available, as they are for some SNL procedures, these measurements should generally be used rather than secondary methods, such as calculation or default values, unless there is some indication or evidence that these measurements are erroneous. If direct measurements of the primary beam itself are unavailable, the ESE can be estimated with a reasonable degree of accuracy from knowledge of the three basic machine settings (applied kilovoltage, current, and time) along with filtration and distance. In addition, collimation, which determines the size of the beam at the worker, is necessary to convert ESE to organ dose.

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<sup>2</sup> This section uses italics to differentiate *exposure* in the special sense from exposure in the general sense. Thus, *exposure* refers to exposure in the special sense. Many publications, including NCRP (1985) and ICRU (1998), discuss exposure in both the general and special sense.

### 3.3.1 Applied Kilovoltage, Filtration, and Beam Quality

The applied kilovoltage refers to the potential between the anode and the cathode of the X-ray tube. In an X-ray tube, electrons released from the cathode are accelerated toward the positively charged anode where they are stopped, giving off most of their kinetic energy in the form of heat, with a small fraction converted to *bremssstrahlung* X-rays with a distribution or spectrum of energies ranging from zero to the applied kilovoltage. Overlying this smooth *bremssstrahlung* spectrum, are peaks from the characteristic K and L X-rays of the tungsten target (anode), which occur at approximately 58 and 11 keV, respectively. For a typical unfiltered X-ray spectrum, the average energy is about one-third of the peak energy or applied kilovoltage. Therefore, most X-rays produced are much lower in energy than the applied kilovoltage of the beam, are attenuated by the torso or other portion of the body, and never reach the film. These X-rays are of little value in radiography but contribute significantly to worker organ dose.

Lower energy X-ray photons can be more or less selectively removed by the addition of a metallic filter in the beam. Filter material is typically of low atomic mass and aluminum is the preferred material for radiographic units. The filter attenuates lower energy photons in the X-ray spectrum more efficiently than higher energy photons; the net effect is to *harden* the beam or increase its quality or average energy and to reduce the entrance exposure to the worker. Beam quality can be expressed in several ways; the most common are in terms of effective energy or half-value layer (HVL). A corollary to filtration is to use a higher applied kilovoltage and to filter the beam relatively heavily to stop most of the low-energy, radiographically useless photons from reaching the worker.

Additional filtration reduces the beam intensity, generally in an exponential manner. For a typical single-phase, half-, full-, or self-rectified machine operating in the range of 80 to 100 kVp, each additional millimeter of aluminum filtration effects a reduction of about 40% in the ESE (Trout, Kelley, and Cathey 1952; Taylor 1957). Therefore, the approximate intensity reduction afforded by any thickness of aluminum filtration can be determined by the following empirical exponential equation:

$$I = I_0 e^{-0.4t} \quad (3-1)$$

or

$$\ln (I/I_0) = -0.4 t \quad (3-2)$$

where:  $t$  = mm Al;

$I$  = beam intensity with filter;

$I_0$  = beam intensity without filter.

In the absence of specific measurements or empirical data, apply this correction to determine the effect of filtration on beam intensity, which is consistent with the guidance in NIOSH (2002a).

All other factors being equal, increasing the kilovoltage will increase the beam intensity or exposure rate. This can be calculated using Kramer's Rule, but such calculations are difficult, complex, and time consuming, and at best results are approximations. However, many empirical studies of beam intensity as a function of kilovoltage provide ample credible evidence to show that, for a given amount of filtration, increasing the applied kilovoltage increases the beam intensity according to the 1.7 power of the applied kilovoltage (Handloser and Love 1951; Trout, Kelley, and Cathey 1952; Kathren 1965; BRH 1970). Other studies and data show a somewhat different power function relationship, but in the absence of specific measurements or empirical data, this function can be applied to determine the effect of applied kilovoltage on beam intensity (see ORAUT 2005a).

### 3.3.2 Current and Exposure Time

Current in an X-ray tube refers to the number of electrons accelerated across the evacuated volume of the tube, which flows from the cathode to the anode; it is therefore proportional to the number of X-rays produced. In theory, for a given applied kilovoltage, the number of X-ray photons produced (and therefore the exposure) will be directly proportional to the X-ray tube current. This is and has been true for most medical radiography units over their design tube current range; it is reasonable and consistent with long-standing radiographic practice to assume linearity of exposure with tube current for a given applied kilovoltage and filtration (Sante 1946).

Exposure time refers to the time the beam is on or the machine is producing X-rays and is, for all practical purposes, linear with exposure. To avoid or minimize image blurring from the beating heart, exposure time for chest radiography was minimized and the current proportionately increased to obtain the desired exposure. However, from a dose reconstruction standpoint, earlier medical radiographic units had mechanical timers with accuracy that was not as good as that of the electronic timers on later machines. Gross bias errors in timer accuracy are unlikely because they would have resulted in over- or underexposure of the radiograph and been quickly detected and corrected. Small random errors that might produce uncertainties of perhaps  $\pm 20\%$  in the exposure are more subtle (see ORAUT 2005a).

X-ray exposures are typically specified in terms of milliampere-seconds (mAs), which is the product of X-ray tube current and exposure time. All other factors remaining constant, the ESE is directly proportional to the number of milliampere-seconds.

### 3.3.3 Distance

X-ray beam intensity is a function of distance from the target, approximating the inverse square at large distances from the tube. Radiographic chest films were taken at a standard source-to-image distance (SID)<sup>3</sup> of 72 in. *Source* refers to the focal spot of the tube, and *image* refers to the plane of the film. The distance to the worker, who was between the source and the film cassette, is sometimes expressed in terms of the source-to-skin distance (SSD) and is somewhat smaller; therefore, the ESE to the worker is somewhat greater than the exposure at the plane of the film. In addition, worker attenuation would further reduce (or attenuate) the number of photons that reached the film. The average worker chest size is 22 to 24 cm (see ORAUT 2005a).

### 3.3.4 Waveform Characteristics

For a given group of settings, waveform characteristics could have an effect on the ESE. X-ray waveforms are of three types: (1) half-wave rectified, which is almost never seen; (2) full-wave rectified, which is typical of medical radiographic units and characteristic of the units at SNL; and (3) constant potential (as defined in NCRP 33). A half-wave-rectified machine produces 60 half-sinusoidal pulses of X-rays per second, each with duration of 1/120 s. A full-wave-rectified machine produces 120 half-sinusoidal pulses per second, each with duration of 1/120 s. Therefore, for a given setting of applied kilovoltage and milliamperes, the intensity of the beam from a half-wave-rectified machine is half that of the beam from the full-wave-rectified machines in use at SNL. A constant potential machine produces a more or less steady (i.e., unpulsed) output of X-rays and has somewhat greater beam intensity (about 10%) than a full-wave-rectified machine operating at the same parameters (see ORAUT 2005a).

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<sup>3</sup> Also known as film-to-focus distance.

### 3.3.5 Collimation

*Collimation* refers to the size of the beam and is of particular importance in the determination of organ doses. The early philosophy in medical radiography was to use a fairly large aperture with limited collimation to ensure that the radiograph included the entire area of interest. With such a limited collimated beam of large size, organs removed from the area of radiographic interest would be in the primary beam and would receive a direct exposure from it. Later, because of protection concerns and to improve beam quality by reducing scatter, beams were collimated such that the smallest beam consistent with the area of interest was used, thereby limiting the exposed area of the worker and, in the case of chest radiography, minimizing dose to organs such as gonads. The use of collimation reduced the dose to the X-ray technologist as well as the worker.

## 3.4 TECHNIQUE FACTORS AND ENTRANCE SKIN EXPOSURE

### 3.4.1 Photofluorography

As no documentation pertaining to PFG apparatus or techniques has been found for Sandia and to ensure favorability to the claimants, a default value of 3.0 cGy (3 rad) for PFG (ORAUT 2005a) is used as the basis for the organ dose in this profile.

### 3.4.2 Radiography, 1953 to 1978

Although the type of apparatus used before 1978 is undocumented, information obtained from an interview with a former SNL X-ray technologist suggests that the same apparatus was in use from startup in 1953 to 1978 (Stout 2005). The technologist, who started work in 1973, indicated that she thought the machine in use at that time was about 20 yr old. She had no recollection of the use of PFG, which is consistent with what the sampling of medical records found. The official SNL history indicates that this X-ray machine went into service in 1953 (SNL 1997). A 1977 radiological protection survey of this unit did not identify machine details, referring to it only as “the medical X-ray unit” (Burnett 1977). The survey report indicated compliance with National Council on Radiological Protection and Measurements (NCRP) Report 33, which recommends a minimum total filter of 2.5 mm Al for operation above 70 kVp (NCRP 1968); the machine was presumably so equipped. The survey report noted a single deficiency (the light field projected by the adjustable collimator was not congruent with the actual X-ray field defined by the collimator) but did not mention the degree of incongruency. The report noted that this condition had been found previously and, because “... no lasting remedy has been found,” recommended continued use of the collimator “because replacement of the complete unit is imminent”; the unit was replaced the following year.

Burnett (1977) included a tabulation of ESE for various types of examinations, including PA chest radiography and AP and LAT lumbar spine. Although stating that these values were, “... several per cent low because no phantom was used in the measurements,” the survey report did not specify measurement technique and instrumentation; therefore, it is not known how the measurements were made; it is therefore impossible to determine if there were errors in measurement. However, based on the reported technique factors, the values cited are inordinately and indeed unrealistically low, even with backscatter correction. For a PA chest radiograph, the ESE is given as 5 mR based on a 70-kVp beam and an exposure of 10 mAs. The *Radiological Health Handbook* indicates that for a 70-kVp beam with a total of 2.5-mm Al filtration, the exposure in mR/mAs would be about 2, giving a more realistic ESE of 20 mR (BRH 1970, pp. 159–160). Approximately the same value is obtained from Table B.3 of NCRP Report 102, which indicates an average air kerma of 0.11 cGy/100 mAs for a 70-kVp beam at the SID of 183 cm (NCRP 1997). For a 10-mAs exposure, this equates to 0.011 cGy or 11 mrad at 183 cm, which converts to 17 mR by the equation

$$D = fR \times (183/152)^2 \quad (3-3)$$

where  $D$  is the air kerma in millirad,  $f$  is a factor equal to 0.93 mrad/mR, and  $R$  is the *exposure* in milliroentgen;  $(183/152)^2$  is the inverse square factor to convert intensity at the SID of 183 cm to intensity at the SSD of 152 cm. To be favorable to the claimant, a suggested default value for the ESE is 20 mR (Table 3-2).

Table 3-2. Machine settings and ESE for screening medical radiography 1953 to 1978.

Procedure	kVp	mAs	Measured ESE (mR) <sup>a</sup>	Calculated ESE (mR)	Comment
PA chest	70	10	5	20	Use calculated ESE as default for dose reconstruction.
AP lumbar spine	75	40	125	400	Use calculated ESE as default for dose reconstruction.
LAT lumbar spine	85	150	600	2,250 <sup>b</sup>	Use calculated ESE as default for dose reconstruction.
AP angle lumbar spine	<sup>c</sup>	<sup>c</sup>	<sup>c</sup>	400 <sup>b</sup>	Use calculated ESE as default for dose reconstruction.
Lumbar spine spot film, LAT	<sup>c</sup>	<sup>c</sup>	<sup>c</sup>	2,250 <sup>b</sup>	Use calculated ESE as default for dose reconstruction.

- Burnett (1977).
- Estimated to ensure favorability to claimants.
- No data are available and the values estimated, as they are not defined.

Similarly, for the AP lumbar spine, Burnett (1977) reports an ESE of 125 mR for an exposure of 40 mAs at 75 kVp and a total filtration of 2.5 mm Al. This measured value seems inordinately low. The reported technique factors (40 mAs at 75 kVp) appear different from those reported in contemporary texts and articles, which indicate an exposure of 50 to 200 mAs at a somewhat lower kVp and SID (not SSD) of 30 to 36 in. (Morgan and Corrigan 1955, p. 388; Sante 1946, 1954; Stanford and Vance 1955), which suggests a significantly higher ESE. The *Radiological Health Handbook* indicates that for the reported technique factors of 40 mAs at 75 kVp and a total filtration of 2.5 mm Al with SSD = 29 in., derived from the standard value of 102 cm given in ICRP 34(1982) corrected for body thickness, the *exposure* would be about 10 mR/mAs, producing a more realistic value of 400 mR for the ESE for an AP lumbar spine radiograph (BRH 1970, p. 159, 160). This is consistent with, and in fact greater than, the ESE of approximately 320 mR obtained from interpolating values listed in Table B.3 of NCRP Report 102 (1997) assuming an exposure of 40 mAs at 75 kVp and SSD = 29 in. and hence, favorable to the claimant.

For a LAT lumbar spine view, Burnett (1977) indicated an ESE of 600 mR for an exposure of 150 mAs at 85 kVp. A more realistic value more favorable to the claimant using these technique factors would be 2,250 mR, based on an SSD of 25 in. and a machine output of 15 mR/mAs interpolated from BRH (1970, p. 159). Interpolation of the values in Table B.3 of NCRP Report 102 (NCRP 1989) gives a somewhat lower albeit consistent ESE of 1,710 mR. To be favorable to the claimants, use the higher value derived from the *Radiological Health Handbook* data in Table 3-2 and the lumbar spine doses are determined and listed in the lumbar spine dose tables.

No data are available for AP angle lumbar spine or LAT lumbar spine films; the calculated ESE values in Table 3-2 are estimates favorable to the claimant based on comparison with standard techniques of the time (Morgan and Corrigan 1955; Sante 1946, 1954).

The values for machine output calculated above and listed in Table 3-2 are based on rounding to ensure favorability to claimants; they are consistent with median values (albeit toward the low side) reported in the contemporary literature (Kereiakes and Rosenstein 1980; Laughlin et al. 1957; Lincoln

and Gupton 1957; Moeller, Terrill, and Ingraham 1953; Wochos, Detorie, and Cameron 1979). In some instances, gonadal doses to males undergoing lumbar spine radiography could have been still lower because the medical records survey indicated that SNL might have used gonadal shields during the 1960s (Stout 2005), which would reduce the gonadal dose significantly. This shield would reduce male gonadal doses by an order of magnitude and if the radiograph indicates the presence of a shield; this is a conservative estimate favorable to the claimant because the dose reduction would probably be much greater than an order of magnitude.

Although SNL radiography procedures are presumed to have been consistent with standard practices of the time and to have followed the adoption of standards of radiology practice during the 1930s and 1940s to minimize dose to the worker, a different impression could be obtained from a 1953 photograph in the official SNL history (SNL 1997), which shows a modern (for 1953) X-ray unit equipped with an adjustable external collimator and beam localization device. The caption reads, "In 1953, two of SNL's five medical doctors were women. Here Dr. Charlotte Beeson and assistant Mary Murphy time an x-ray on a patient in the medical department's new x-ray room." The photograph is obviously posed because it is inconsistent with standard medical X-ray practice of the time, with the documented low doses incurred by X-ray technologists, and with the radiological protection survey in 1977 (Burnett 1977). It shows a worker with her back to the photographer lying on her right side on what appears to be a standard simple X-ray table. The X-ray tube is approximately 3 ft above her. Dr. Beeson stands at the head of the table apparently adjusting the position of the worker's head. Ms. Murphy stands about 30 in. from the head of the table holding the mechanical timer in both hands. Between Dr. Beeson and Ms. Murphy at the head of the table is the X-ray control panel that was probably wheeled to this location for the photograph.

### 3.4.3 Radiography, 1978 to 1996

In 1978, the SNL medical department installed a new Picker GX 325 X-ray unit. This unit was equipped with a phototimer that was used for most chest X-rays and automatic collimation to limit the beam to the appropriate film size. On August 28, 1978, Herb Abbott performed a survey of the machine and reported an ESE of 20 mR for a chest X-ray taken at 104 kVp and 100 mA with an exposure time of 0.1 s (10 mAs), 180 mR for an AP lumbar spine taken at 75 kVp and 100 mA with an exposure time of 0.5 s (50 mAs), and 920 mR for a LAT lumbar spine taken at 90 kVp and 100 mA with an exposure time of 1.5 s (150 mAs) (Stout 2005). These are indicative of high-voltage techniques, the use of which was just beginning, that greatly reduced worker dose. The reported ESE values are fully consistent with what would be expected with the use of these techniques. Moreover, they are internally consistent. Although neither the SSD nor the SID is specified in Abbott's notes (Stout 2005), the observed ratios of ESEs for the various procedures are entirely consistent with what would be expected using standard SID distances (chest SID = 72 in., lumbar spine SID = 36 in.), as can be demonstrated with the aid of the sample calculation below using the relationships described in Section 3.2:

$$R = (d_2/d_1)^2 \times (mAs_1/mAs_2) \times (kVp_1/kVp_2)^{1.7} \quad (3-4)$$

where:

$R$  = the ratio of ESEs for the two procedures, in this case LAT lumbar spine (920 mR) and PA chest (20 mR), respectively;

$(d_2/d_1)^2$  = the distance correction factor obtained by inverse square using SID = 36 in. for the lumbar spine and 72 in.;

$mAs_1/mAs_2$  = the exposure correction factor, which is the ratio of mAs for the two procedures;

$(kVp_1/kVp_2)^{1.7}$  = the correction for the differing kVp used in the two procedures.

Putting in the indicated values yields:

$$(72/36)^2 \times (150/10) \times (90/104)^{1.7} = 4 \times 15 \times 0.78 = 46.8$$

which is very close to the ratio of  $920/20 = 46$  measured by Abbott. Note that the above calculation is based on SID rather than SSD. The measured and calculated ratios are within two percent of each other and calculated ratios are well within the expected uncertainty as discussed in Section 3.6. A similar calculation can be done to compare the measured and calculated ratios between the AP lumbar spine view and the PA chest. In this case, the measured ratio is  $180/20 = 9$  and the calculated ratio is  $4 \times 5 \times 0.57 = 11.4$ , some 27 per cent greater, but still within the expected uncertainty.

Radiation safety surveys conducted on January 29, 1991, and May 4, 1993, by the Public Health Service revealed no operational problems or deficiencies (Payne 1991, 1993). The survey reports did not include beam measurements, which presumably were not made. A January 1991 study of estimated skin doses on a phantom for various X-ray procedures reported results for shallow and deep doses (Sanderville 1991). Shallow dose was assumed to be the skin dose including backscatter and was converted to ESE by use of backscatter factors from NCRP Report 102 (NCRP 1997, Table B.8). For the mid-PA chest, taken at 120 kVp with an exposure of 3.3 mAs, a dose of 21 mrem was reported. Using a backscatter factor of 1.4 interpolated from NCRP Report 102, assuming 1 mrem = 1 mrad, and using a conversion factor of 0.93 mrad/mR, results in an ESE of 16 mR, which is fully consistent with values calculated from or reported in the contemporary literature. Table 3-3 lists similar values calculated for the lumbar spine views using a backscatter factor favorable to the claimant of 1.3 along with the measured shallow dose values.

Table 3-3. Machine settings and ESE for screening radiography, 1978 to 1996.

Procedure	kVp	mAs	Measured shallow dose <sup>a</sup> (mrem)	Calculated ESE (mR)
PA chest	120	3.3	21	16
AP lumbar spine	75	50	385	318
LAT lumbar spine	80	80	754	750
AP angle lumbar spine	<sup>b</sup>	<sup>b</sup>	385	318
LAT lumbar spine spot	90	80	954	750

a. Includes backscatter.

b. No data are available.

#### 3.4.4 Radiography, 1997 to Present

In 1996, SNL changed the X-ray apparatus and installed a Picker HF 500 unit. This machine was equipped with automatic exposure at 3 mAs that was used for most chest radiographs and automatic collimation to limit the beam size to that of the film used. Most exposures were made with the phototimer. Table 3-4 lists manual machine settings and corresponding calculated ESE values.

Table 3-4. Manual settings and ESEs for screening medical radiography, 1997 to present.

Procedure	KVp	Exposure (mAs)	SID (in.)	Calculated ESE <sup>a</sup> (mR)
PA chest	120	3.3	72	20

AP lumbar spine	76	40	40	450
LAT lumbar spine	90	50	40	750

a. Calculated from measurements from Heintz (2005) and rounded.

On March 13, 1997, this apparatus was evaluated as part of a comprehensive radiation protection survey performed by Robert G. Antonsen, Jr., CSO, a radiation specialist with the U.S. Food and Drug Administration (Antonsen 1997). According to the report, the X-ray unit met all applicable standards and specifications with the exception that the acceptable standard total percent difference (length and width) between the X-ray field and light field alignment for the wall cassette of 4% was slightly exceeded (4.4%) when the automatic collimation was evaluated (Antonsen 1997). Correction was made 11 d after the survey report was received by SNL (Stout 1997). This small deviation should have had no significant impact on organ doses.

An in-depth radiographic unit performance summary by Philip H. Heintz, a Board-certified medical physicist, on May 20, 2005, found that the unit was in compliance with the New Mexico Radiation Protection Code and met current performance recommendations of the American Association of Physicists in Medicine (Heintz 2005). Beam quality at 80 kVp was determined and a calculated HVL of 3.71 mm Al was obtained. Thus, for dose reconstruction purposes, the authors assume an HVL of 4 mm Al to account for higher kVp techniques and to be favorable to the claimant. Machine output measurement data in terms of mR/mAs were given for various applied kilovoltages, which enabled calculation of the ESE for various procedures. Table 3-4 summarizes these for various radiographic procedures and manual technique factors.

ESE values were calculated using the beam output measurements reported in Heintz (2005) with suitable corrections for distance and kVp. The ESE values in Table 3-4, which are based on measured values reported by Heintz, are significantly higher than those measured by Antonsen (1997) for chest (13 mR) and lumbar spine (357 mR) with manual techniques. The reasons for this discrepancy are not readily apparent; because the reports do not provide full details on the exposure parameters, it is impossible to draw other than speculative conclusions. It is reasonable to assume that Antonsen used the technique factors listed in Table 3-4 because these have not changed since machine installation in 1996. However, the Antonsen report does not specify which lumbar spine view was measured and it does not give the distance. A significant part of the discrepancy for the chest film could be accounted for if Antonsen was reporting *exposure* at the SID rather than the SSD. In any case, dose reconstructors should use ESE values calculated from the reported measurements of Heintz (2005) for organ dose calculations; these values are favorable to the claimant because exposures made using manual settings will typically be about 30% to 35% greater than exposures made using the phototimer for chest and lumbar spine radiography.

### 3.5 X-RAY DOSES TO WORKERS

Extensive review of the available documentation on the occupational medical program at SNL from 1953 to the present revealed the following required medical radiographic procedures administered in connection with preemployment or regular postemployment medical examinations:

- PA 14- by 17-in. chest
- Photofluorographic 4- by 5-in. chest
- AP lumbar spine
- LAT lumbar spine
- AP angle of S1
- LAT spot film, L5-S1

Although other radiographic examinations of workers could have occurred as a result of illness or injury, there is no indication in the records that SNL routinely performed other radiographic examinations or treatments, such as those performed in the past for shrinkage of lymphoid tissue, on workers as a condition of employment [3]. Therefore, only doses from the six techniques listed above were evaluated.

### 3.5.1 Conversion of ESE to Dose

X-ray intensity or the output of an X-ray machine has typically been measured in terms of *exposure* and units of roentgen. Over the years, the definition of *exposure* has undergone a number of changes. There have also been changes in the values of the conversion factors used to convert *exposure* to absorbed dose; at various times an *exposure* of 1 R would be equated to a soft tissue dose of 0.83, 0.877, or 0.93 rad. Thus, an *exposure* of 1 R would result in an absorbed dose of somewhat less than 1 rad (1 cGy = 10 mGy). DOE Orders and those of its predecessor agencies have defined 1 R as exactly equal to a dose of 1 rad (10 mGy), thereby producing a small overestimate in the reported dose or dose equivalent because dosimeters were typically calibrated against a field measured in roentgen, which was numerically equated as absorbed dose in rad (Kathren and Petersen 1989). Further complicating the conversion of ESE in terms of exposure to absorbed dose is the contemporary trend to refer to X-ray intensity in terms of the quantity *kerma*, which is measured in the same units as absorbed dose. The numerical value of kerma is typically slightly lower than the corresponding value of absorbed dose. Therefore, to ensure conservatism and compliance with NIOSH (2002a) and to avoid any risk of dose underestimation, 1 R of exposure was taken to be equal to 1 rad of absorbed dose in soft tissue and to 1 rad (10 mGy) of kerma.

Conversion of exposure expressed as ESE to organ dose is determined using the published conversion factors in Tables A.2 to A.8 of International Commission on Radiological Protection (ICRP) Publication 34 (ICRP 1982). These tables list average absorbed organ doses for selected medical radiography procedures related to an entrance air kerma without backscatter of 1 Gy for various beam qualities expressed in terms of HVL of aluminum. However, the tables in ICRP Publication 34 do not list all organs identified in the NIOSH Interactive RadioEpidemiological Program (IREP) software. For organs in IREP but not specifically identified in ICRP Publication 34, use of the dose conversion factors for the organ identified in ICRP Publication 34 that is anatomically the closest is a reasonable and simple first-order approach that generally would be favorable to the claimant or neutral. Therefore, the factor for lung would be applied to all other organs in the thoracic cavity (thymus, esophagus, liver, gall bladder, spleen, and stomach) as listed in Table 3-5. Because an appreciable fraction of the skeleton (in particular the trabecular bone that has a large surface-to-volume ratio and the sternum that is a primary location of red marrow in the adult) is in the trunk, the factor for lung would also be applied to the bone surfaces and remainder organs. For organs in the abdomen (i.e., urinary bladder and colon/rectum), the dose conversion factor for ovary would be used. For the eye/brain, the analogue organ is the thyroid. The doses for IREP organs that are not listed in ICRP Publication 34 were determined using the analogue organs listed in Table 3-5.

Table 3-5. Analogues for IREP organs not listed in ICRP Publication 34.

Anatomical location	ICRP 34 reference organ	IREP organ analogues
Thorax	Lung	Thymus Esophagus Stomach Bone surface Liver, gall bladder, spleen Remainder organs

Abdomen	Ovaries	Urinary bladder Colon/rectum
Head and neck	Thyroid	Eye/brain

Because 1 R is taken to be exactly equal to 10 mGy of kerma, conversion from ESE to organ dose could be made easily if the beam quality was known. With the exception of the most recent SNL X-ray machine, measured beam quality data were not found. However, the applied kilovoltage and filtration are known and an estimate of beam quality could be made from these data. For a given amount of filtration and exposure (mAs), absorbed organ dose increases as a function of HVL; so for conservatism, the upper limit on the likely beam quality was calculated and rounded up to match the closest value in the tables in ICRP Publication 34 (ICRP 1982). For the period before 1996, beam quality expressed as HVL was conservatively estimated to be 2.5 mm Al; after 1996, the estimate was 4 mm Al based on the measurements by Heintz (2005). These values are somewhat greater than the 1.75-mm Al values that would be derived from Table A.16 of ICRP Publication 34 and, therefore, are favorable to the claimant.

### 3.5.2 Organ Doses from Chest Photofluorography

Although no documentation pertaining to PFG apparatus or techniques has been found, a sampling of the radiographs in employee medical files indicates that most routine chest films taken from the inception of the medical program in 1953 until 1966 appear to have been obtained by PFG [3]. Because no measurements or other information on PFG apparatus have been found and, in accordance with the guidance in ORAUT (2005a), the default entrance kerma of 3.0 cGy (3 rad) is used as the basis for organ doses in this site profile and is favorable to the claimants.

Organ dose calculations used the exposure expressed as ESE with dose conversion factors (DCF) in Tables A.2 to A.8 of ICRP Publication 34 (ICRP 1982). These tables list organ doses based on an SID of 183 cm for chest radiography. PFG used an SID of 40 in. or 102 cm. Given the geometric considerations in relation to divergence of the beam as discussed in ICRP Publication 34 (pp. 23ff), this difference in distance would have a negligible effect on organ doses and, therefore, the values listed in Tables A.2 to A.8 are appropriate for chest PFG. Because absorbed organ doses for radiographs are a function of beam quality, an HVL of 2.5 mm Al was used for dose determination based on an assumed 70 kVp (typical for chest PFG) and 2.5-mm Al filtration.

The basic calculational methodology is to convert ESE to entrance kerma. As discussed above, an exact numerical relationship is assumed to be favorable to the claimant and, therefore, 1 R = 1 rad = 10 mGy. Therefore, for the default ESE of 3 R, the corresponding entrance kerma value is 3 rad or 30 mGy. This value is used with the DCFs in Tables A.2 to A.8 of ICRP Publication 34 (ICRP 1982). For example, Table A.2 would be used to obtain the dose to the thyroid. The DCF for a PA chest for a beam quality of 2.5 mm Al is listed as 32 mGy/1 Gy entrance kerma. The dose for a 30-mGy entrance kerma can be calculated by simple proportionality. First, the DCF is converted to units of mGy:

$$\text{thyroid dose} = 32 \text{ mGy per entrance kerma Gy} = 32 \text{ mGy}/1,000 \text{ mGy}$$

then multiplied by the entrance kerma, also in units of mGy:

$$\text{thyroid dose} = 32 \text{ mGy}/1,000 \text{ mGy} \times 30 \text{ mGy entrance kerma} = 0.960 \text{ mGy}$$

The same basic procedure is used for all organ dose calculations. The dose in gray can be converted to dose equivalent in units of rem by multiplying by 100, a factor derived from the conversion 1 Gy = 100 rad and a radiation weighting factor of unity to convert absorbed dose to dose equivalent.

Table 3-6 lists calculated organ doses for chest PFG using the default ESE of 3 R. As noted in the footnotes to the table, doses to ovaries, testes, and their analogues were scaled up from actual measurements made at the DOE Hanford Site, where the ESE was 1.53 R (Rising and Soldat 1959).

Table 3-6. Organ dose estimates for SNL chest PFG, 1953 to 1966.

Organ	View	Organ dose (rem) <sup>a</sup>
Thyroid	PA-PFG	5.2E-01
Eye/brain	PA-PFG	9.6E-02
Ovaries <sup>b</sup>	PA-PFG	2.5E-02
Urinary bladder	PA-PFG	2.5E-02
Colon/rectum	PA-PFG	2.5E-02
Testes <sup>b</sup>	PA-PFG	5.0E-03
Lungs	PA-PFG	1.4
Thymus	PA-PFG	1.4
Esophagus	PA-PFG	1.4
Stomach	PA-PFG	1.4
Bone surface	PA-PFG	1.4
Liver/gall bladder/spleen	PA-PFG	1.4
Remainder	PA-PFG	1.4
Breast	PA-PFG	0.15
Uterus/embryo	PA-PFG	2.5E-02
Bone marrow	PA-PFG	2.76E-01
Skin <sup>c</sup>	PA-PFG	4.1

a. Rem values are from ORAUT (2005a).

b. Modified ovaries and testes doses from Webster and Merrill (1957); use actual measurement data if available.

c. Skin dose was determined by multiplying the ESE by the backscatter factor of 1.35 (for HVL of 2.5 mm Al) from NCRP (1997, Table B.8).

### 3.5.3 Organ Doses from 14-in. by 17-in. PA Chest Radiography

The dominant X-ray procedure at SNL was chest PFG until 1966, when it was discontinued and apparently replaced by 14- by 17-in. PA chest radiography but in some cases, 14 x 17 films were performed in addition to the PFG or alone without the PFG. Organ doses for PA chest films were calculated using the exposure expressed as ESE with DCFs in Tables A.2 to A.8 of ICRP (1982). Table 3-7 summarizes the ESE, entrance kerma, and beam quality values used for organ dose calculations. Table 3-8 lists the DCFs used for the organ dose calculations derived from ICRP 34 (1982) and Table 3-9 lists calculated doses for all IREP organs.

### 3.5.4 Organ Doses from Lumbar Spine Radiography

Lumbar spine radiographs were routinely required for certain classes of male workers to determine the presence of back problems; selection was based on medical history and job classification [4]. The frequency of lumbar spine views, if required, was variable. If required as a condition of employment,

Table 3-7. ESE, entrance kerma, and HVL for 14- by 17-in. PA chest radiography.

Period	ESE (mR)	Entrance kerma (mGy)	Beam quality (HVL in mm Al)
1953–1978	20	0.20	2.5
1979–1995	16	0.16	2.5
1996–present	20	0.20	4.0

Table 3-8. DCFs for 14- by 17-in. PA chest radiography from ICRP Publication 34.<sup>a</sup>

Organ	View	SID (cm)	Image receptor size (cm)	DCF (mGy per Gy air kerma)		
				(Beam quality 2.5 mm Al HVL) minimal collimation 1953 to 1970	(Beam quality 2.5 mm Al HVL) 1971 to 1996	(Beam quality 4.0 mm Al HVL) after 1996
Thyroid	PA	183	35.6 × 43.2	174 <sup>c</sup>	32	78
Ovaries	PA	183	35.6 × 43.2	168 <sup>b</sup>	1	5.2
Testes	PA	183	35.6 × 43.2	9.1 <sup>b</sup>	<0.01	<0.01
Lungs (male)	PA	183	35.6 × 43.2	419	419	628
Lungs (female)	PA	183	35.6 × 43.2	451	451	674
Breast (female)	PA	183	35.6 × 43.2	49	49	116
Uterus/embryo	PA	183	35.6 × 43.2	149 <sup>b</sup>	1.3	5.2
Bone marrow (male)	PA	183	35.6 × 43.2	92	92	178
Bone marrow (female)	PA	183	35.6 × 43.2	86	86	172

a. DCFs from Tables A.2 to A.8 in ICRP (1982). Analogue organs not listed in table.

b. DCFs are based on analogue organs (abdominal), are very conservative on the high side, and are favorable to the claimant.

c. Value obtained from AP Cervical spine DCF corrected for depth by a factor of 0.2 from ORAUT (2005).

Table 3-9. Organ dose estimates for 14- by 17-in. PA chest radiography.

Organ	Minimal collimation 1953–1970	1971–1996 dose (rem)	1996–present dose (rem)
Thyroid	3.5E-03	5.1E-04	1.6E-03
Eye/brain	3.5E-03	5.1E-04	1.6E-03
Ovaries	3.4E-03	1.6E-05	1.0E-04
Urinary/bladder	3.4E-03	1.6E-05	1.0E-04
Colon/rectum	3.4E-03	1.6E-05	1.0E-04
Testes	1.8E-04	1.6E-07	2.0E-07
Lungs (male)	8.4E-03	6.7E-03	1.3E-02
Lungs (female)	9.0E-03	7.2E-03	1.3E-02
Thymus	9.0E-03	7.2E-03	1.3E-02
Esophagus	9.0E-03	7.2E-03	1.3E-02
Stomach	9.0E-03	7.2E-03	1.3E-02
Bone surface	9.0E-03	7.2E-03	1.3E-02
Remainder organs	9.0E-03	7.2E-03	1.3E-02
Breast	9.8E-04	7.8E-04	2.3E-03
Uterus/embryo	3.0E-03	2.1E-05	1.0E-04
Bone marrow (male)	1.8E-03	1.5E-03	3.6E-03
Bone marrow (female)	1.7E-03	1.4E-03	3.4E-03
Skin <sup>a</sup>	2.7E-02	2.2E-02	2.8E-02

a. Skin dose was determined by multiplying the ESE by the backscatter factors of 1.35 and 1.4 for HVL of 2.5 mm Al and 4.0 mm Al from NCRP (1997, Table B.8), respectively.

lumbar spine radiographs were typically performed as part of the preemployment/new hire examination. This could have been the only occasion on which many workers received lumbar spine radiographs. However, the possibility of periodic lumbar spine worker examinations, including an exit employment physical examination, should not be precluded. The initial number of persons selected for the lumbar spine series was relatively large, but was only a small fraction of the total number of employees. By 1985, required lumbar spine radiography reportedly ceased (Stout 2005). The procedure could have persisted after 1985, but there is no documentation or anecdotal evidence to this effect.

Lumbar spine examinations for evaluating back problems could have included as many as four views: (1) AP, (2) LAT, (3) AP angle at the S1 level, and (4) L5-S1 LAT spot film. Table 3-10 lists previously

determined ESE values for all four lumbar spine radiographic procedures. Table 3-11 lists DCFs from ICRP (1982). Table 3-12 lists organ dose estimates for AP lumbar spine and AP angle, S-1 level projections. Table 3-13 lists organ dose estimates for LAT lumbar spine and LAT spot film, L5-S1 level projections. If projections 1 and 2 and projections 3 and 4 listed above are provided to the worker, double the dose to the organs as the doses in Table 3-12 and 3-13 are for a single projection. Recommended practice was to use a 5-in. cone for improved radiographic quality (Sante 1954, p. 207), which limited the beam diameter to 5 in. at the skin entrance point, but it is not known if lumbar spine projections were coned views, which would have resulted in lower organ doses than an open or un-coned procedure. Therefore, to be favorable to the claimant, it was assumed that all projections were taken without cones; this is reflected in the organ dose estimates in Tables 3-12 and 3-13.

Table 3-10. ESE for lumbar spine radiography.

Projection	1953–1978 ESE (mR)	1979–1985 ESE (mR)
AP lumbar spine	400	318
LAT lumbar spine	2,250	750
AP angle, S1 level	400	318
LAT spot film, L5-S1 level	2,250	750

Table 3-11. DCFs for lumbar spine radiography from ICRP (1982).<sup>a</sup>

Organ	Source-image distance (cm)	Image receptor size (cm)	DCF (mGy per Gy air kerma) (Beam quality 2.5 mm aluminum HVL)	
			AP and AP angle	LAT and spot
Thyroid	102	35.6 × 43.2	0.3	<0.01
Ovaries	102	35.6 × 43.2	216	47
Testes	102	35.6 × 43.2	4.2	0.8
Lungs	102	35.6 × 43.2	79	14
Breast (female)	102	35.6 × 43.2	79 <sup>b</sup>	14 <sup>b</sup>
Uterus/embryo	102	35.6 × 43.2	287	31
Bone marrow	102	35.6 × 43.2	37	22

a. DCFs from ICRP (1982, Tables A.2 to A.8). Analogue organs not listed in table.

b. DCF not given in ICRP 34(1982) but noted as small. To be favorable to the claimant, use the DCF for lungs.

### 3.6 UNCERTAINTY ANALYSIS

*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, provide an indication of the confidence or validity of the dose estimates. Error implies knowledge of the correct or actual value, which is, of course, not known. Therefore, the more appropriate factor is uncertainty, which is expressed in terms of a confidence level, which in turn is expressed as a percent. Therefore, the 99% confidence level indicates that the correct or true value, although not actually known, has a 99% probability of falling within the range

Table 3-12. Organ dose estimates for AP lumbar spine and AP angle spine, S1 radiography.<sup>b</sup> The doses are from a single projection. Double the doses for the total dose to the worker if both projections are given.

Organ	AP projection 1953–1978 dose (rem)	AP projection 1979–1985 dose (rem)
Thyroid	1.2E-04	9.5E-05
Eye/brain	1.2E-04	9.5E-05

Ovaries	8.5E-02	6.9E-02
Urinary/bladder	8.5E-02	6.9E-02
Colon/rectum	8.5E-02	6.9E-02
Testes	1.7E-03	1.3E-03
Lungs	3.1E-02	2.5E-02
Thymus	3.1E-02	2.5E-02
Esophagus	3.1E-02	2.5E-02
Stomach	3.1E-02	2.5E-02
Bone surface	3.1E-02	2.5E-02
Remainder organs	3.1E-02	2.5E-02
Breast	3.1E-02	2.5E-02
Uterus/embryo	0.11	9.1E-02
Bone marrow	1.5E-02	1.2E-02
Skin <sup>a</sup>	0.540	0.429

- Skin dose was determined by multiplying the ESE by the backscatter factor of 1.35 for HVL of 2.5 mm Al per NCRP (1997), Table B.8.
- Double the dose if both the AP lumbar spine and AP angle spine, S1 exams are given.

cited. The statement of confidence level typically includes all potential sources of error (random and systematic); the precision or reproducibility of the measurement; and accuracy, or how close the measurement or estimate of dose comes to the actual or correct value.

While in theory, a large number of factors can introduce uncertainties or affect the X-ray machine output intensity and dose to the worker, in practice, only the following five factors can be reasonably considered to have a meaningful or significant impact on dose uncertainty:

- Measurement error
- Variation in applied kilovoltage (kVp)
- Variation in beam current (mA)
- Variation in exposure time (s)
- Distance from the worker to the source of the X-rays (SSD)

The influence of other factors such as use of screens, grids, reciprocity failure, film speed, and development, while potentially variable, do not affect the beam output intensity *per se* except indirectly insofar as these can determine the exposure settings (kVp, mA, and time) used.

### 3.6.1 Measurement Error

Beam intensity from a medical X-ray unit was largely derived from actual measurement of X-ray machine output with R meters or similar ionization chamber devices designed for measurement of photons in the medical X-ray energy range. If properly calibrated and used, such instruments typically

Table 3-13. Organ dose estimates for LAT lumbar spine and LAT lumbar spot film, L-5 radiography.<sup>b</sup> The doses are from a single projection. Double the doses for the total dose to the worker if both projections are given.

Organ	LAT Projections 1953–1978 dose (rem)	LAT Projections 1979–1985 dose (rem)
Thyroid	2.3E-05	7.5E-06
Eye/brain	2.3E-05	7.5E-06

Ovaries	1.1E-01	3.5E-02
Urinary/bladder	1.1E-01	3.5E-02
Colon/rectum	1.1E-01	3.5E-02
Testes	1.8E-03	6.0E-04
Lungs	3.2E-02	1.1E-02
Thymus	3.2E-02	1.1E-02
Esophagus	3.2E-02	1.1E-02
Stomach	3.2E-02	1.1E-02
Bone surface	3.2E-02	1.1E-02
Remainder organs	3.2E-02	1.1E-02
Breast	3.2E-02	1.1E-02
Uterus/embryo	7.0E-02	2.3E-02
Bone marrow	5.0E-02	1.7E-02
Skin <sup>a</sup>	3.04	1.013

- a. Skin dose was determined by multiplying the ESE by the backscatter factor of 1.35 for HVL of 2.5 mm Al per NCRP (1997, Table B.8).
- b. Double the dose if both LAT lumbar spine and LAT lumbar spot film, L-5 exams are given.

and historically have had an uncertainty of  $\pm 2\%$  for photon energies below 400 keV (Kathren and Larson 1969). Although more recent versions of these instruments might provide a somewhat smaller uncertainty, perhaps on the order of  $\pm 1\%$  (NBS 1985, 1988), dose reconstructors should apply the uncertainty range of  $\pm 2\%$  to measurements of X-ray intensity for conservatism.

X-ray measurements were sometimes made with instruments not designed for the medical X-ray energy range, which exhibited pronounced energy dependence. Use of such instruments could result in large errors, sometimes as much as an order of magnitude or more; dose reconstructors should discard measurements made with inappropriate instruments.

### 3.6.2 Variation in Applied Kilovoltage and Beam Current

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, as can be the case if the power rating of the machine is exceeded, it is unlikely that power ratings were ever exceeded because such an event would be difficult to achieve in practice and could result in damage to the X-ray tube, which would necessitate replacement. However, even with the use of constant voltage transformers to control line voltages, slight variations could occur in line voltage input or other internal voltages, which in turn could alter the applied kilovoltage of the output beam. In general, for a given applied kilovoltage setting, variation in the applied kilovoltage falls within  $\pm 5\%$  or less of the machine setting, as noted with the current SNL apparatus (Antonsen 1997; Payne 1991; Heintz 2005). Beam intensity is approximately proportional to the 1.7 power of the applied kilovoltage; this translates to an uncertainty of approximately  $\pm 8.6\%$  in relation to output beam intensity in the 80- to 100-kVp range used for chest radiographs, which is rounded up to  $\pm 9\%$  for conservatism.

Slight variations in tube current are similarly normal; as a tube ages or heats from use, current can change and typically will drop. With all other factors constant, beam intensity will be reduced in direct proportion to the change in tube current. The reduction in beam output from current variation is typically not more than a few percent under normal operating conditions and generally goes undetected because such a small change has minimal impact on radiograph quality. Large decreases are readily detectable, manifest themselves as underexposed radiographs, and result in maintenance on the machine to restore the output or, as a temporary measure, an increase in the current or applied

kilovoltage to provide the necessary intensity for proper radiography. For a given applied kilovoltage setting, the output of the beam is a function of the tube current, which is measured by a milliammeter. The measurement is subject to uncertainties; there can be minor changes in output as the tube heats from normal use. These variations are typically small and the estimated uncertainty in beam intensity or output attributable to current variation is  $\pm 5\%$ .

### **3.6.3 Variation in Exposure Time**

Another parameter that could affect the dose from a radiograph, perhaps significantly, is the time of exposure. The potential importance of this parameter is underscored by the fact that virtually all medical X-ray units in the DOE Complex were of the full-wave-rectified type. A full-wave-rectified machine produces 120 pulses/s. In a typical radiographic exposure time of 1/20 s, only six pulses would result. A small error in the timer that resulted in a change of only  $\pm 1$  pulse would affect the output by  $\pm 17\%$ ; for an exposure time of 1/30 s, the change in output corresponding to a deviation of 1 pulse is  $\pm 25\%$ . Early mechanical timers, such as that shown in the 1953 photograph of the SNL X-ray apparatus, were notoriously inaccurate; accuracy improved significantly with the introduction of electronic timers. Measurements of reproducibility made in the late 1980s and beyond by the State of Washington for machines at the Hanford Site suggest that the timers, and indeed the entire X-ray output, were fairly constant (ORAUT 2005a). It is not always known whether mechanical or electronic timers were used with manual radiographic operations; therefore, for conservatism, the assumed uncertainty in beam output attributable to timers is  $\pm 25\%$ .

### **3.6.4 Distance Uncertainty**

The final factor likely to affect worker dose relates to distance from the source of the X-rays, which is an important determinant of the ESE from which organ doses are calculationally derived. For a given individual, the SSD will be determined largely by the body thickness of the worker and the accuracy of the positioning. For a typical worker, the estimated variation in SSD is no more than a few centimeters, with an upper limit of perhaps 7.5 cm. Because exposure follows the inverse square law, this indicates an uncertainty of  $\pm 10\%$  from this source.

### **3.6.5 Combined Total Uncertainty**

There are two approaches to determine the combined uncertainty from the five potential sources of dose uncertainty listed above. 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  $2 + 9 + 5 + 25 + 10 = \pm 51\%$ . A more reasonable approach would be to assume that the uncertainties are random, and to compute the statistical root mean square (RMS) value. The RMS value is the square root of the sum of the squares and computes as  $\pm 28.9\%$ . Rounding this up to  $\pm 30\%$  would seem to provide an adequate and suitably conservative indication of uncertainty. Therefore, for an individual ESE or derived organ dose, dose reconstructors should assume an uncertainty of  $\pm 30\%$ . For further conservatism, it might be appropriate to assume that errors are all positive and that only  $+30\%$  should be used.

## **4.0 OCCUPATIONAL ENVIRONMENTAL DOSE**

### **4.1 INTRODUCTION**

#### **4.1.1 Purpose**

This section provides a technical basis for evaluating the occupational environmental dose for EEOICPA claimants who were employed at SNL-NM and the TTR in Nevada. Occupational

environmental dose refers to radiation exposures received by workers while on the site but outside the Sandia National Laboratories (SNL) facilities from facility discharges to the atmosphere, from ambient external radiation originating in the facilities, and from inadvertent ingestion of site-generated radionuclides. The receptors of concern are SNL employees who did not wear external dosimetry or who were not monitored for internal exposures.

The SNL-NM facility includes the five Technical Areas (TAs) south of Albuquerque, New Mexico, on the Kirtland Air Force Base. The TTR is 140 miles northwest of Las Vegas, Nevada.

#### **4.1.2 Scope**

This section describes environmental occupational exposures at SNL facilities but outside the occupational workplace. It includes estimated annual intakes of radionuclides from inhalation and inadvertent ingestion and the estimated radiation dose from ambient external exposures from 1948 to 2004. Results for 2005 are currently being prepared by SNL staff members and will be available in the 2005 Annual Site Environmental Reports for SNL-NM and SNL-NV. That information will be incorporated when it becomes available. Estimates for different facilities vary by year and by type of exposure because sources of radionuclides and radiation did not exist at all facilities for all years during this period.

### **4.2 OPERATIONS OVERVIEW**

#### **4.2.1 Sandia National Laboratories – New Mexico**

SNL-NM began as the Albuquerque branch of Los Alamos Scientific Laboratory (LASL) and operated from 1945 to 1949. Sandia Corporation was formed in 1949 and the name was changed to Sandia Laboratories in 1971 and to SNL in 1979. Although a single comprehensive history of Sandia facility radiological operations has not been identified by the authors, information on construction and operation of new facilities can be gathered from annual environmental monitoring reports as well as from a few existing historical summaries (Ulrich 1998; SNL 2006). The layout of the SNL-NM site as of 2005 is shown in Figure 4-1.

TA-I began operations in 1946, and temporary buildings were used until construction of more permanent buildings for an expanded facility began in 1948. The Albuquerque Laboratory site concentrated primarily on development and testing of nonnuclear components. Activities included use of a Cockroft-Walton accelerator and Van de Graaff generator, as well as a set of chemistry laboratories that used radiochemicals to support research needs. Environmental monitoring reports have noted the potential for release of tritium from TA-I (Millard, Gray, and O'Neal 1984), but often note that no releases occurred. A Neutron Generator Facility was completed in 1996 and has since been the source of larger releases of tritium.

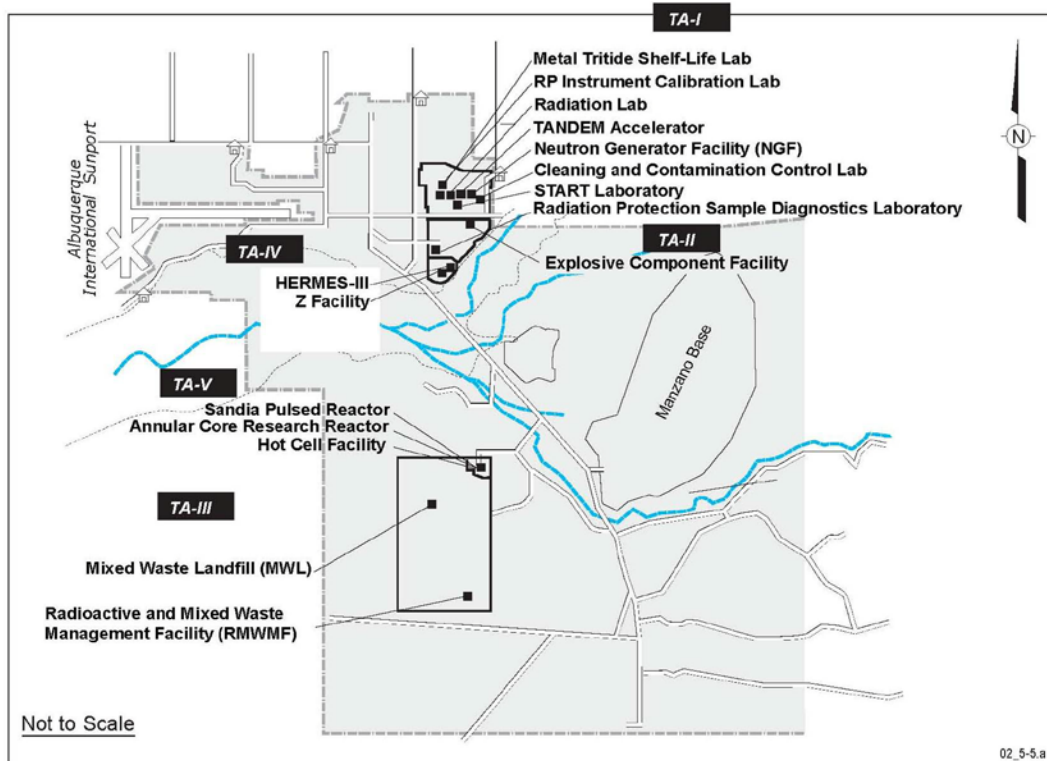


Figure 4-1. SNL-NM site south of Albuquerque, New Mexico, in 2005.

Early manufacturing activities at Sandia Laboratories included the assembly of weapons. This resulted in the development of the separate TA-II about a half mile south of TA-I, for the handling and incorporation of explosives into the weapons. Construction of TA-II, which paralleled the development of TA-I, was initiated in 1948. The resultant construction included two identical assembly buildings and a control building completed in 1949. Assembly activities continued at TA-II until 1957. The records note the potential for the release of small amounts of  $^{85}\text{Kr}$  vented from the earth by explosive testing during fracture permeation tests (Millard, Gray, and O'Neal 1984).

The increasing use of missiles as delivery vehicles led to the full-scale environmental testing of weapons with and without explosives. As a result, there was a need for complex equipment and specialized engineers to analyze the test results. A decision was made to centralize a group of test devices in TA-III approximately 7 miles south of TA-I. Planning for this area began in 1952 and the first group of facilities, which consisted of a centrifuge, a rocket-sled, a vibration testing facility, and an instrument control center, was completed in 1953. The Mixed Waste Landfill (MWL) was operated at TA-III from 1959 to 1988 as a disposal site for low-level radioactive and mixed waste. The Radioactive and Mixed Waste Management Facility (RMWMF) was completed in 1995 for repackaging low-level radioactive and transuranic wastes from current and legacy activities at SNL and its predecessors.

In 1957, a nuclear reactor facility was proposed for what was to become TA-V. TA-V is in the extreme northeast corner of TA-III and is remote from TA-I, TA-II, and TA-IV as well as from most of the facilities in TA-III. The Sandia Pulse Reactor (SPR) facility was started in May of 1961. The SPR is an unreflected, cylindrical, enriched-uranium assembly. Small amounts of fission product gases and air activation products (primarily  $^{41}\text{Ar}$ ) are produced as released during operations. The Sandia Engineering Reactor (SER) (or SERF) was a 5-MW heterogeneous reactor fueled with aluminum-clad  $^{235}\text{U}$ -enriched U-Al alloy elements. The reactor was cooled and moderated by light water and was operational from October 1962 to June 1969. Air activation products (primarily  $^{41}\text{Ar}$ ) were released

through the stack. The Annular Core Pulse Reactor (ACPR) (now the Annular Core Research Reactor [ACRR]) is a modified Training, Research, Isotopes General Atomics (TRIGA) reactor that became operational in June of 1967. Air activation products (primarily  $^{41}\text{Ar}$ ) were released through the stack (Brewer 1973). A hot cell facility began operation in 1979 (Simmons 1979, 1980). Other facilities in TA-V have included electron beam accelerators, a GIF, and a neutron irradiation facility. Tritium could have been released from the accelerators during operations and other radionuclides could have been released from hot cell operations.

Beginning in 1979, TA-IV was developed as a location for inertial confinement fusion research and pulsed power research. The first facility was PBFA-I (originally to be called the Electron Beam Fusion Facility), which was completed in 1980 with an upgrade to PBFA-II completed in 1985. After additional upgrades, this accelerator is now called the Z Pinch Machine. Gaseous tritium (HT) effluents were noted to be generated from TA-IV activities during some years (Millard, Gray, and O'Neal 1984), but not during others. More accelerator facilities were subsequently developed in TA-IV, including HERMES and others. These accelerators have or had the potential to generate short-lived activation product radionuclides as well as external radiation.

#### 4.2.2 Sandia National Laboratories – Nevada

The TTR is about 140 miles northwest of Las Vegas, Nevada, and 32 miles southeast of Tonopah, Nevada. It is on approximately 526 square miles at the northern boundary of the Nevada Test and Training Range (NTTR; formerly Nellis Air Force Range), which was established in 1940 by President Roosevelt as the Las Vegas Bombing and Gunnery Range. The topography at TTR is characterized by a broad, flat, valley bordered by two north- and south-trending mountain ranges: (1) the Cactus Range to the west (occurring mostly within the boundaries of TTR) and (2) the Kawich Range to the east. Cactus Flat is the valley floor where the main operational area of TTR is located. An area of low hills outcrops in the south. Elevations within TTR range from 1,630 m (5,347 ft) at the valley floor to 2,279 m (7,482 ft) at Cactus Peak. The elevation within the town of Tonopah is at 1,837 m (6,030 ft) (Sanchez, Hamilton, and Mayeux 2001a). Figure 4-2 shows the layout of SNL-NV at TTR.

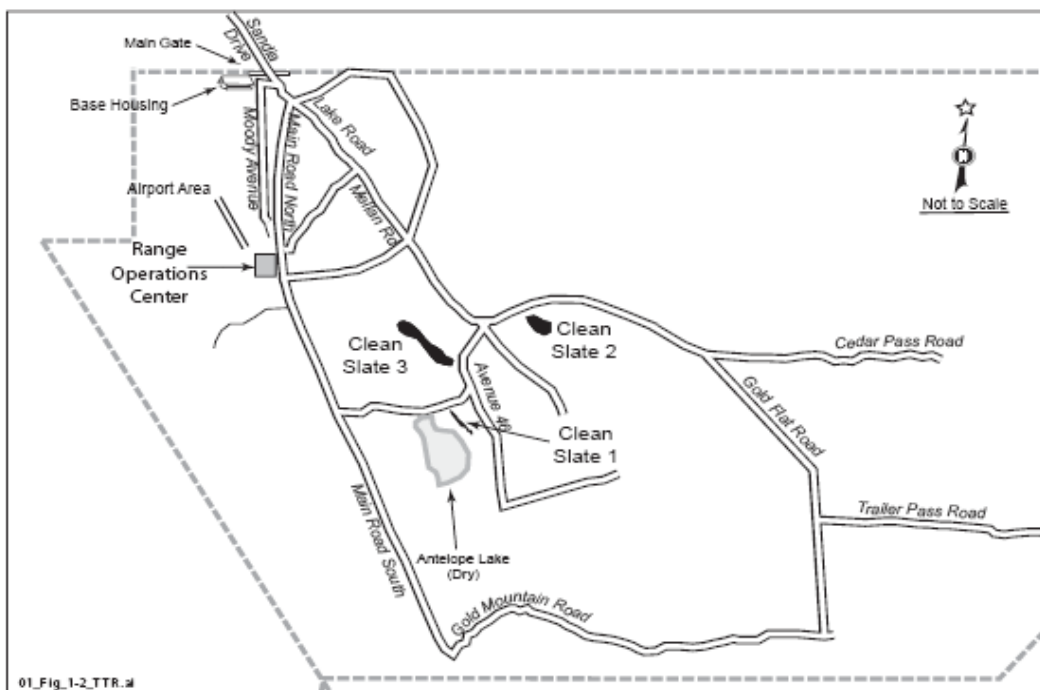


Figure 4-2. Layout and location of facilities operated by SNL-NV at TTR in 2004.

The TTR was eventually selected as a bombing range after similar facilities at the Salton Sea Test Base in California, as well as Yucca Flat on the NTS, became inadequate. By the mid-1950s, the atmosphere at the Salton Sea Test Base became permeated with haze, which limited visibility and hampered photography. NTS's Yucca Flat site also became inadequate due to the increasing emphasis on low-altitude approaches and deliveries that required flat terrain and a long approach corridor. The TTR site, which was approximately seven times the size of the Salton Sea Test Base, was well suited because it had immense areas of flat terrain needed for the increasing use of rockets and low-altitude, high-speed aircraft operations (Sanchez, Hamilton, and Mayeux 2001a).

The TTR area was withdrawn in 1956 and Sandia began activities in 1957 to operate and test new weapon systems. In the years following World War II, facilities that were built at TTR were originally designed and equipped to gather data on aircraft-delivered inert test vehicles under U.S. Atomic Energy Commission (AEC) cognizance (now U.S. Department of Energy [DOE]). Over the years, the facilities and capabilities at TTR were expanded to accommodate tests related to the DOE Weapons Ordnance Program (Sanchez, Hamilton, and Mayeux 2001a).

The Main Compound in Area 3 is the heart of the test range activities. The Operations Control Center controls and coordinates all test functions and affords a 360° view of the site. During test operations, the test director, range safety officer, test project engineer, camera controller, and range communicator operate the consoles in the Operations Control Center to control and coordinate all test functions. Another important location at the range is Area 9, which has weapons storage facilities and is used to conduct ground-to-air rocket launching tests (Sanchez, Hamilton, and Mayeux 2001a).

Principal DOE activities at TTR include stockpile reliability testing; research and development testing support of structural development; arming, fusing, and firing systems testing; and testing nuclear weapon delivery systems. However, no nuclear devices are tested at TTR. TTR is instrumented with a wide array of signal-tracking equipment including video; high-speed cameras; radar tracking devices used to characterize ballistics, aerodynamics, and parachute performance on artillery shells; bomb drops; missiles; and rockets (Sanchez, Hamilton, and Mayeux 2001a).

In recent years, specific test activities at TTR have consisted of the following:

- Air drops (trajectory studies of simulated weapons)
- Gun firings
- Ground-launched rockets (study of aeroballistics and material properties)
- Air-launched rockets (deployed from aircraft)
- Explosive testing (e.g., shipping and storage containers)
- Static rocket tests (related to the Trident Submarine Program)
- Ground penetrator tests

These activities require a remote range both for public safety and to maintain national security. The majority of test activities at TTR occur within Cactus Flat, a valley with almost no topographical relief flanked by mountains and hills (Sanchez, Hamilton, and Mayeux 2001a).

### **Clean Slate and Double Tracks Sites**

In May and June 1963, Project Roller Coaster included a series of four nuclear weapons destruction tests that resulted in plutonium dispersal in the surrounding soils. These tests were conducted to study plutonium dispersal from accidental nonnuclear explosions of plutonium-bearing weapons (Millard and Lathrop 1982). Three of these tests were conducted within the boundaries of TTR; the fourth was conducted on the NTTR just west of TTR. The three Project Roller Coaster test sites at TTR are referred to as Clean Slates 1, 2, and 3 (as shown in Figure 4-2). The fourth test site at NTTR

is referred to as Double Tracks. In 1996, the Double Tracks was closed after remediation of soil contamination was reduced to a level of less than or equal to 200 pCi/g of transuranic radionuclides (Sanchez, Hamilton, and Mayeux 2001a).

The initial cleanup of each Clean Slate site was conducted shortly after each test. Test-related debris was bladed into a hole at ground zero and backfilled. An initial fence was built around each test area where the soil contamination was set at approximately 1,000  $\mu\text{g}/\text{m}^2$  of plutonium. The soil survey was conducted on 61-m grids with a hand-held survey meter or field instrument for the detection of low-energy radiation. In 1973, additional outer fences were set at 40 pCi/g of plutonium in soil also using the hand-held meter method. Soil sampling is conducted periodically at these sites and the areas are visually inspected twice a year to determine whether fence repairs are required. Horses that wander inside the fenced areas are promptly removed (Sanchez, Hamilton, and Mayeux 2001a).

The Clean Slate sites are the only known sources of potential occupational environmental radionuclide exposure at TTR.

### 4.3 RADIONUCLIDE SCREENING, SNL-NEW MEXICO

SNL-NM's principle research activities have mainly been nonradiological; these activities have never been large producers of radionuclides as have other DOE production operations. Radiation and radionuclides have been used to support research activities; their use has generally increased since Sandia's early days. Many of the research activities have been episodic, with durations of several years before being suspended or changed.

Radioactive effluent monitoring information for SNL-NM was available for 1973 to 2004 and included listed airborne emissions of 69 radionuclides. Many of these radionuclides were released in very small quantities that were likely to be negligible contributors to worker dose. Therefore, a radionuclide screening was conducted to identify those radionuclides that were the most important contributors to dose. The parameters for the screening were as follows:

- The maximum yearly release for each radionuclide was evaluated.
- The screening threshold was set at 0.01 mrem/yr for the maximum release.
- The exposure pathways of inhalation and external dose from plume passage were considered, using the air dose screening factors from NCRP Report 123 (NCRP 1996).
- The radionuclides were assumed to be released only during the normal working day (over a period of 2,000 hr rather than continuously over 8,760 hr); workers were assumed to be exposed for 2,000 hr/yr.
- The atmospheric dispersion factor was assumed to be 0.001  $\text{s}/\text{m}^3$ , a value approximately 3 times higher than the highest atmospheric dispersion factor for a ground-level release at 100 m calculated using Albuquerque-specific meteorology from CAP88PC Version 3.0 (EPA 2006).

The screening indicated 21 radionuclides with the potential to exceed the 0.01-mrem/yr threshold for the maximum yearly release. These radionuclides and their exposure pathways are shown in Table 4-1.

Table 4-1. Radionuclides meeting screening criteria and the principle worker exposure pathway.

Inhalation	External plume	Both pathways
Am-241, H-3, I-129, I-131, I-133, U-238, Th-232, Co-60, Sr-90	Ar-41, Kr-85m, Kr-87, N-13, O-15, Xe-133, Xe-133m, Xe-135, Xe-135m	I-135, Rb-88, Kr-88

#### 4.4 AMBIENT EXTERNAL RADIATION

##### 4.4.1 Sandia National Laboratories – New Mexico

Ambient external radiation dose information for SNL-NM is available for 1980 to 2004 (Simmons 1980; Millard et al. 1981, 1982, 1983, 1987, 1988, 1989a; Millard, Gray, and O’Neal 1984; Millard, Gray, and Thompson 1985, 1986; Hwang et al. 1990a, 1991a; Culp et al. 1992, 1993, 1994; Shyr et al. 1995; Shyr, Duncan, and Sanchez 1996; Fink, Duncan, and Sanchez 1997, 1998, 1999; Duncan and Sanchez 2001; Sanchez, Hamilton, and Mayeux 2001b; Sanchez et al. 2002a; Wagner et al. 2003a, 2004a, 2005a). Sources of ambient external radiation could have been direct external radiation produced by reactors in TA-V and accelerators (principally those in TA-IV) and exposure to external radiation from plumes of short-lived activation products released from accelerator operations. Evaluation of the potential external doses from these plumes showed potential doses to be very low in most cases, on the order of a few microrem per year ( $\mu\text{rem/yr}$ ) using the screening criteria in Section 4.3.

Starting in 1980, thermoluminescent dosimeters (TLDs) were used to monitor the environmental ionizing radiation background levels in areas surrounding the research facilities at SNL, in areas along the perimeter of Kirtland Air Force Base, and within and surrounding the city of Albuquerque. Field TLDs were used in sets of five and were replaced and measured quarterly. TLDs were prepared and evaluated by the SNL Dosimetry Laboratory of the Health Instrumentation Division. The results of the TLD measurements were reported in annual environmental monitoring reports to DOE.

The locations of the environmental TLD stations were specified by the Environmental Protection and Hazardous Waste Management Division and described in the annual environmental monitoring reports. The TLD data were reported in millirem per year from 1980 to 1998, and in milliroentgen per year from 2000 to 2004. From 1980 to 1985 and from 1993 to 1999, only the arithmetic mean plus one standard deviation was reported. The TLD data for each station were reported for 1986 to 1992 and for 2000 to 2004. The total systematic error for the environmental TLD field results was reported to be  $-1.3 \pm 7.0\%$  (bias  $\pm$  uncertainty). The most significant contributions to total systematic error were directional dependence ( $-1.4\%$  bias) and energy response (5.5% uncertainty). Outliers were removed from the data before analysis based on a residuals analysis of the number of observations and level of significance. The number of outliers removed was limited to one-fifth for each subgroup.

Examination of TLD results shows little difference between onsite and offsite locations except in several locations where area dosimeters were known to be located in controlled areas with higher external radiation dose rates. There are infrequent instances of unusually high quarterly results for TA-V. Activities in TA-V were most likely to be associated with external doses to area workers. An assumption favorable to the claimant was made that workers in these areas could have received external doses in the range of 10 mrem/yr during the years these areas were active. It was assumed that workers in TA-III adjacent to TA-V could also have received some lower exposure from TA-V activities. Accelerator activities in TA-IV were assumed to have the potential to result in higher external doses and workers in TA-I and TA-II could have received some lower exposure from TA-IV activities. These estimates of external dose to unmonitored workers in the different TAs are based on a working year of 2,000 hr and are shown in Table 4-2. These doses include direct external radiation and exposure to external radiation from passing plumes.

Table 4-2. External radiation dose for workers in SNL-NM TAs (mrem/yr).<sup>a</sup>

Year	TA-I	TA-II	TA-III	TA-IV	TA-V
1948–1952	< 1	< 1	N/A	N/A	N/A
1953	< 1	< 1	< 1	N/A	N/A
1954	< 1	< 1	< 1	N/A	N/A
1955	< 1	< 1	< 1	N/A	N/A
1956	< 1	< 1	< 1	N/A	N/A
1957	< 1	< 1	< 1	N/A	N/A
1958	< 1	< 1	< 1	N/A	N/A
1959	< 1	< 1	< 1	N/A	N/A
1960	< 1	< 1	< 1	N/A	N/A
1961	< 1	< 1	5	N/A	10
1962	< 1	< 1	5	N/A	10
1963	< 1	< 1	5	N/A	10
1964	< 1	< 1	5	N/A	10
1965	< 1	< 1	5	N/A	10
1966	< 1	< 1	5	N/A	10
1967	< 1	< 1	5	N/A	10
1968	< 1	< 1	5	N/A	10
1969	< 1	< 1	5	N/A	10
1970	< 1	< 1	5	N/A	10
1971	< 1	< 1	5	N/A	10
1972	< 1	< 1	5	N/A	10
1973	< 1	< 1	5	N/A	10
1974	< 1	< 1	5	N/A	10
1975	< 1	< 1	5	N/A	10
1976	< 1	< 1	5	N/A	10
1977	< 1	< 1	5	N/A	10
1978	< 1	< 1	5	N/A	10
1979	< 1	< 1	5	N/A	10
1980	5	5	5	10	10
1981	5	5	5	10	10
1982	5	5	5	10	10
1983	5	5	5	10	10
1984	5	5	5	10	10
1985	5	5	5	10	10
1986	5	5	5	10	10
1987	5	5	5	10	10
1988	5	5	5	10	10
1989	5	5	5	10	10
1990	5	5	5	10	10
1991	5	5	5	10	10
1992	5	5	5	10	10
1993	5	5	5	10	10
1994	5	5	5	10	10
1995	5	5	5	10	10
1996	5	5	5	10	10
1997	5	5	5	10	10
1998	5	5	5	10	10
1999	5	5	5	10	10
2000	5	5	5	10	10
2001	5	5	5	10	10
2002	5	5	5	10	10
2003	5	5	5	10	10
2004	5	5	5	10	10

a. Based on a working year of 2,000 hr.

#### 4.4.2 Sandia National Laboratories – Nevada

The U.S. Environmental Protection Agency (EPA) has monitored external radiation at TTR and at Goldfield and Tonopah, Nevada, since at least 1971 (EPA 1972) as part of overall NTS offsite environmental monitoring. These early results typically include a single dosimeter reading for the three locations. EPA results are reported in annual environmental monitoring reports for the TTR (Millard and Lathrop 1982, 1984, 1985; Millard 1986; Millard and West 1987, 1988; Millard et al. 1989b; Hwang et al. 1990b, 1991b; Howard and Culp 1992; Culp and Howard 1993; Culp, Howard and McClellan 1994; Culp and Forston 1995, 1996; Culp et al. 1997, 1998; Duncan and Sanchez 1999; Duncan, Forston, and Sanchez 2000; Sanchez, Hamilton, and Mayeux 2001b; Sanchez et al. 2002b; Wagner et al. 2003b; 2004b, 2005b) as well as in the EPA offsite environmental monitoring reports. These results do not show significant differences in external radiation dose between the NTS offsite locations and TTR.

A separate long-term environmental external radiation monitoring network using TLDs at TTR was established by SNL-NV in January 1994. Environmental TLDs were placed at various locations off the site, at the site perimeter, and on the site to measure gamma radiation (Culp and Forston 1995). This program significantly expanded the level of external radiation monitoring at TTR. In general, these results also do not show a significant difference in external radiation dose from the offsite locations. However, there is one dosimeter location (T-13) at the northeast corner of the Operations Center perimeter fence that had elevated results for 1995, 1996, and 1997. During these years, the average annual dose at T-13 was  $210 \pm 26$ ,  $260 \pm 42$ , and  $218 \pm 52$  mrem, respectively, while the site annual averages including these results were  $144 \pm 22$ ,  $159 \pm 17$ , and  $150 \pm 21$  mrem. Results were reported for only two of three periods in 1994, but the third period record was  $108.1 \pm 9.9$  mrem, which was higher than would be expected and consistent with the higher results of the following years. Soil sampling (OC-1) was conducted near T-13 and the results did not indicate radionuclide contamination. Gamma surveys were performed using hand-held radiation survey equipment; the surveys found no unusual results. Reviews of the historic TLD data for T-13 showed the elevated results did not occur every monitoring period but were more cyclic or intermittent in nature (Culp and Forston et al. 1998).

The possible source of these higher TLD results was never identified in the annual monitoring reports. However, a teleconference with one of the authors of the annual reports indicated he had later gained information that these higher results could have related to radiography of test assemblies in the Operations area (Ikenberry 2006a). A teleconference with the Project Lead for Radiation Protection Dosimetry at SNL indicated that all workers had personal TLDs and external dosimetry records were available for all workers since 1990 (Ikenberry 2006b). Therefore, the potential for radiation exposure from radiography operations would be included in the workers' personal external dosimetry records. Furthermore, the Project Lead was of the opinion that this trend was established well before 1990, but that only the 1990 and later records were readily and electronically available from the dosimetry database. Previous radiation exposures for each individual were lumped together in a pre-1990 total external dose record in the database for each worker; however, individual year records should be available on hardcopy. Table 4-3 shows estimates of external dose to unmonitored workers are zero for all years of TTR operations.

Table 4-3. External radiation dose for workers at TTR (mrem/yr).

Years	Annual external radiation dose
1957–2004	0

## 4.5 INHALATION OF ONSITE AIRBORNE RADIONUCLIDES

### 4.5.1 Sandia National Laboratories – New Mexico

The environmental monitoring program at SNL-NM was initiated in December 1958 to establish background radiation levels before the startup of the reactors in TA-V (Burnett et al. 1961). There is no indication of radionuclide release before TA-V reactor activities; however, assumptions favorable to the claimant about earlier releases may be made for the TAs.

Radionuclide air emission data in curies per year are available in the annual environmental monitoring reports for SNL-NM from 1973 to 2004 (Brewer 1974; Holley 1975; Holley and Simmons 1976; Simmons 1977, 1978, 1979, 1980; Millard et al. 1981, 1982, 1983; Millard, Gray, and O'Neal 1984; Millard, Gray, and Thompson 1985, 1986; Millard et al. 1987, 1988, 1989a; Hwang et al. 1990a, 1991a; Culp et al. 1992, 1993, 1994; Shyr et al. 1995; Shyr, Duncan, and Sanchez 1996; Fink, Duncan, and Sanchez 1997, 1998, 1999; Duncan and Sanchez 2001; Sanchez, Hamilton, and Mayeux 2001b; Sanchez et al. 2002a; Wagner et al. 2003a, 2004a, 2005a). The 1973 report states that since SNL releases only small amounts of  $^{41}\text{Ar}$  from TA-V and tritium from TA-I as a result of operations, the requested dose estimates are based on effluent release data rather than environmental monitoring data.  $X/Q$  values were calculated for the various distances from the release points to the site boundaries. In 1979,  $^{85}\text{Kr}$  was included as an effluent from TA-II. In 1989,  $^{133}\text{Xe}$ ,  $^{13}\text{N}$ ,  $^{15}\text{O}$ , and  $^{238}\text{U}$  were added to the list of radioactive air emissions. In 1990,  $^{129}\text{I}$  was reported as a release from the HCF at TA-5.

Three findings from the 1991 Tiger Team Assessment of SNL-NM questioned the adequacy of airborne radionuclide emission monitoring. Sandia responded (SNL 1992a) that although SNL did not have any point source with the potential for doses to the public greater than 0.1 mrem/yr that would require continuous monitoring, it had installed stack monitors capable of continuous monitoring at two facilities in TA-V: (1) the ACRR and (2) the Hot Cell Facility (HCF). As a result of this assessment, the number of reported effluents from SNL increased in 1991 to 32 radionuclides released from 13 facilities. This trend of reporting more radionuclides from more facilities continued through 2004.

Reported facility radionuclide emissions were the basis for estimating potential worker environmental inhalation intakes. Years with data vary by facility, but the most complete information is available for 1991 and later. For years where there were no emission data, the facility-specific average for available years was typically used, especially if multiple years of data were available. If the radionuclide releases appeared episodic for a specific type of campaign or research, then the first year of effluent data might have been used to characterize earlier years without data.

Identification of specific radionuclides released from various facilities in stack emissions are available in data from 1973 to 2004 and were used to characterize radionuclide emissions for all years. There are no radionuclide-specific data from earlier years. Any uranium and thorium detected is assumed to originate from facility activities, even though it could be naturally occurring.

Estimates of inhalation intakes were made using the exposure criteria that were included in the initial radionuclide screening. These criteria are:

- The radionuclides were assumed to be released only during the normal working day (over a period of 2,000 hr rather than continuously over 8,760 hr), which maximized potential worker exposure; workers were also assumed to be exposed for 2,000 hr/yr.

- The atmospheric dispersion factor was assumed to be  $0.001 \text{ s/m}^3$ , a value approximately 3 times higher than the highest atmospheric dispersion factor for a ground-level release at 100 m calculated using Albuquerque-specific meteorology from CAP88PC Version 3.0 (EPA 2006).
- In addition, the worker breathing rate for light activity was  $1.5 \text{ m}^3/\text{h}$  ( $4.2 \times 10^{-4} \text{ m}^3/\text{s}$ ) (ICRP 1994a).

Use of these parameters will result in estimates of radionuclide inhalation intakes that are favorable to the claimant.

#### 4.5.1.1 Technical Areas I, II, and IV

No emission monitoring data were available from before 1973 for TA-I and TA-II, but information in early environmental monitoring reports indicates that the potential for radionuclide release was minimal (Burnett et al. 1961; Brewer 1973). TA-IV was not created until 1979. Environmental monitoring data indicate that tritium ( $^3\text{H}$ ) was the principle radionuclide released; most available data was from 1991 and later. The Building 805 radiation laboratory was the source of small releases of  $^{241}\text{Am}$  and  $^{238}\text{U}$  with data available for a limited number of years.

The annual average release of tritium from years with emission monitoring data was assumed to apply to all years without data, back to the earliest years of the laboratories when permanent buildings were erected (1948). This assumption is favorable to the claimant because it will likely overestimate the inhalation of tritium for TA-I and TA-II. In addition, the release of small quantities of  $^{241}\text{Am}$  and  $^{238}\text{U}$  was assumed to apply to all years of operations for TA-I and TA-II. This is also an assumption favorable to the claimant that will likely overestimate inhalation of  $^{241}\text{Am}$  and  $^{238}\text{U}$ . The only radionuclide released in TA-IV that could contribute to inhalation dose was tritium. Inhalation of tritium was assumed to be the same as inhalation in TA-I and TA-II from 1979 and later. This is also an assumption favorable to the claimant that will likely overestimate inhalation of tritium in TA-IV. The estimated inhalation intakes of radionuclides for TA-I, TA-II, and TA-IV are shown in Table 4-4. The material form and solubility should be selected for each radionuclide based on the values that are most favorable to the claimant.

Table 4-4. SNL-NM TAs-I, -II, and -IV maximum annual median intakes (Bq/yr) via inhalation.<sup>a</sup>

Year	Radionuclide			
	TA-I and TA-II			TA-IV
	H-3	Am-241	U-238	H-3
1948–1952	3E+03	2E-07	6E-08	N/A
1953	3E+03	2E-07	6E-08	N/A
1954	3E+03	2E-07	6E-08	N/A
1955	3E+03	2E-07	6E-08	N/A
1956	3E+03	2E-07	6E-08	N/A
1957	3E+03	2E-07	6E-08	N/A
1958	3E+03	2E-07	6E-08	N/A
1959	3E+03	2E-07	6E-08	N/A
1960	3E+03	2E-07	6E-08	N/A
1961	3E+03	2E-07	6E-08	N/A
1962	3E+03	2E-07	6E-08	N/A
1963	3E+03	2E-07	6E-08	N/A
1964	3E+03	2E-07	6E-08	N/A
1965	3E+03	2E-07	6E-08	N/A

Year	Radionuclide			
	TA-I and TA-II			TA-IV
	H-3	Am-241	U-238	H-3
1966	3E+03	2E-07	6E-08	N/A
1967	3E+03	2E-07	6E-08	N/A
1968	3E+03	2E-07	6E-08	N/A
1969	3E+03	2E-07	6E-08	N/A
1970	3E+03	2E-07	6E-08	N/A
1971	3E+03	2E-07	6E-08	N/A
1972	3E+03	2E-07	6E-08	N/A
1973	2E+03	2E-07	6E-08	N/A
1974	7E+02	2E-07	6E-08	N/A
1975	8E+02	2E-07	6E-08	N/A
1976	7E+02	2E-07	6E-08	N/A
1977	2E+02	2E-07	6E-08	N/A
1978	6E+04	2E-07	6E-08	N/A
1979	7E+04	2E-07	6E-08	N/A
1980	1E+00	2E-07	6E-08	1E+00
1981	1E+00	2E-07	6E-08	1E+00
1982	1E+00	2E-07	6E-08	1E+00
1983	1E+00	2E-07	6E-08	1E+00
1984	1E+00	2E-07	6E-08	1E+00
1985	1E+00	2E-07	6E-08	1E+00
1986	1E+00	2E-07	6E-08	1E+00
1987	1E+00	2E-07	6E-08	1E+00
1988	1E+00	2E-07	6E-08	1E+00
1989	1E+00	2E-07	6E-08	1E+00
1990	4E+00	2E-07	6E-08	4E+00
1991	1E+00	2E-07	6E-08	1E+00
1992	2E+00	2E-07	6E-08	2E+00
1993	2E+00	2E-07	6E-08	2E+00
1994	2E+00	2E-07	6E-08	2E+00
1995	8E+00	2E-07	6E-08	8E+00
1996	3E+03	7E-09	6E-08	3E+03
1997	5E+04	7E-09	6E-08	5E+04
1998	2E+05	7E-09	6E-08	2E+05
1999	6E+04	7E-09	6E-08	6E+04
2000	8E+04	7E-09	6E-08	8E+04
2001	1E+05	7E-09	6E-08	1E+05
2002	8E+03	7E-09	6E-08	8E+03
2003	1E+04	7E-09	6E-08	1E+04
2004	3E+03	7E-09	6E-08	3E+03

- a. Radionuclide solubility class should be selected to be favorable to the claimant.

#### 4.5.1.2 Technical Area III

No emission monitoring data were available from before 1973 for TA-III, but information in early environmental monitoring reports indicates that the potential for radionuclide release was minimal (Burnett et al. 1961; Brewer 1973). TA-III activities began in 1953; it is unclear if any radionuclides were involved during the earliest years. Environmental monitoring data indicate that tritium ( $^3\text{H}$ ) was the principle radionuclide released, mainly from the diffuse source MWL with data available for 1993 and later. The Chemical Waste Landfill (CWL) has been a diffuse source of small quantities of  $^{60}\text{Co}$ ,

$^{232}\text{Th}$ , and  $^{238}\text{U}$ . Building 6600 was a small source of  $^{241}\text{Am}$  in the mid-1990s and the RMWMF (Building 6920) was a comparatively large source of  $^{241}\text{Am}$  and  $^{90}\text{Sr}$  from 2002 to 2004.

The annual average release of tritium from years with emissions monitoring data was assumed to apply to all years without data since 1953 when TA-III activities first began. This is an assumption favorable to the claimant that will likely overestimate the inhalation of tritium for TA-III.

The release of small quantities of  $^{232}\text{Th}$ ,  $^{238}\text{U}$ , and  $^{60}\text{Co}$  from the CWL was assumed to apply to all years of operations for TA-III. This is also an assumption favorable to the claimant that will likely overestimate inhalation of these radionuclides. There was no radioactive decay correction of  $^{60}\text{Co}$  to increase inhalation intakes in earlier years. Release and inhalation of  $^{241}\text{Am}$  and  $^{90}\text{Sr}$  from the RMWMF was assumed to occur only during the years in which these radionuclides were identified (2002 to 2004). The RMWMF has been operational only since 1995 and has been continuously monitored during that time. Releases of very small amounts of  $^{241}\text{Am}$  from Building 6600 were assumed to occur during the specific years emissions were detected during the 1990s, although releases were also assumed to occur during the period from 1999 to 2001 to provide continuity between Building 6600 and RMWMF  $^{241}\text{Am}$  releases.

The estimated inhalation intakes of radionuclides for TA-III are shown in Table 4-5. The material form and solubility should be selected for each radionuclide based on the values most favorable to the claimant.

Table 4-5. SNL-NM TA-III maximum annual median intakes (Bq/yr) via inhalation.<sup>a</sup>

Year	Radionuclide					
	H-3	Am-241	U-238	Th-232	Co-60	Sr-90
1953	5E+04	N/A	8E-02	2E-04	4E+00	N/A
1954	5E+04	N/A	8E-02	2E-04	4E+00	N/A
1955	5E+04	N/A	8E-02	2E-04	4E+00	N/A
1956	5E+04	N/A	8E-02	2E-04	4E+00	N/A
1957	5E+04	N/A	8E-02	2E-04	4E+00	N/A
1958	5E+04	N/A	8E-02	2E-04	4E+00	N/A
1959	5E+04	N/A	8E-02	2E-04	4E+00	N/A
1960	5E+04	N/A	8E-02	2E-04	4E+00	N/A
1961	5E+04	N/A	8E-02	2E-04	4E+00	N/A
1962	5E+04	N/A	8E-02	2E-04	4E+00	N/A
1963	5E+04	N/A	8E-02	2E-04	4E+00	N/A
1964	5E+04	N/A	8E-02	2E-04	4E+00	N/A
1965	5E+04	N/A	8E-02	2E-04	4E+00	N/A
1966	5E+04	N/A	8E-02	2E-04	4E+00	N/A
1967	5E+04	N/A	8E-02	2E-04	4E+00	N/A
1968	5E+04	N/A	8E-02	2E-04	4E+00	N/A
1969	5E+04	N/A	8E-02	2E-04	4E+00	N/A
1970	5E+04	N/A	8E-02	2E-04	4E+00	N/A
1971	5E+04	N/A	8E-02	2E-04	4E+00	N/A
1972	5E+04	N/A	8E-02	2E-04	4E+00	N/A
1973	5E+04	N/A	8E-02	2E-04	4E+00	N/A
1974	5E+04	N/A	8E-02	2E-04	4E+00	N/A
1975	5E+04	N/A	8E-02	2E-04	4E+00	N/A
1976	5E+04	N/A	8E-02	2E-04	4E+00	N/A
1977	5E+04	N/A	8E-02	2E-04	4E+00	N/A
1978	5E+04	N/A	8E-02	2E-04	4E+00	N/A
1979	5E+04	N/A	8E-02	2E-04	4E+00	N/A

Year	Radionuclide					
	H-3	Am-241	U-238	Th-232	Co-60	Sr-90
1980	5E+04	N/A	8E-02	2E-04	4E+00	N/A
1981	5E+04	N/A	8E-02	2E-04	4E+00	N/A
1982	5E+04	N/A	8E-02	2E-04	4E+00	N/A
1983	5E+04	N/A	8E-02	2E-04	4E+00	N/A
1984	5E+04	N/A	8E-02	2E-04	4E+00	N/A
1985	5E+04	N/A	8E-02	2E-04	4E+00	N/A
1986	5E+04	N/A	8E-02	2E-04	4E+00	N/A
1987	5E+04	N/A	8E-02	2E-04	4E+00	N/A
1988	5E+04	N/A	8E-02	2E-04	4E+00	N/A
1989	5E+04	N/A	2E-01	2E-04	4E+00	N/A
1990	5E+02	N/A	8E-02	2E-04	4E+00	N/A
1991	5E+02	N/A	8E-02	2E-04	4E+00	N/A
1992	1E+03	N/A	8E-02	2E-04	4E+00	N/A
1993	4E+04	N/A	8E-02	2E-04	4E+00	N/A
1994	7E+03	2E-09	8E-02	2E-04	4E+00	6E-03
1995	7E+03	2E-09	8E-02	2E-04	4E+00	6E-03
1996	1E+05	2E-09	8E-02	2E-04	4E+00	6E-03
1997	6E+04	2E-09	8E-02	2E-04	4E+00	6E-03
1998	2E+04	2E-09	2E-03	2E-04	4E+00	6E-03
1999	2E+04	2E-09	8E-02	2E-04	4E+00	6E-03
2000	2E+04	2E-09	8E-02	2E-04	1E+01	6E-03
2001	7E+03	2E-09	8E-02	2E-04	3E-04	6E-03
2002	1E+04	4E-03	8E-02	2E-04	3E-01	6E-03
2003	4E+05	2E-01	8E-02	2E-04	4E+00	3E-01
2004	3E+04	2E-01	8E-02	2E-04	4E+00	6E-03

- a. Radionuclide solubility class should be selected to be favorable to the claimant.

#### 4.5.1.3 Technical Area V

No emission monitoring data were available before 1973 for TA-V, but information in early environmental monitoring reports indicates that the potential for radionuclide release was minimal (Burnett et al. 1961; Brewer 1973). TA-V activities began in 1961 with operation of the SPR. Early environmental monitoring reports indicate that most TA-V releases were noble gases (addressed in Section 4.4). There is some indication of tritium releases in the late 1970s from TA-V and for some small releases of radioiodines and other radionuclides from the ACRR and HCF in the mid-1990s.

The annual average release of tritium from years with emission monitoring data was assumed to apply to all years without data back to 1961 when TA-V activities first began. This is an assumption favorable to the claimant that will likely overestimate the inhalation of tritium for TA-III.

The release of small quantities of  $^{129}\text{I}$ ,  $^{131}\text{I}$ ,  $^{133}\text{I}$ ,  $^{135}\text{I}$ ,  $^{88}\text{Kr}$ , and  $^{88}\text{Rb}$  were assumed to apply to all years because the ACRR began operation in 1967. Some of the releases of these radionuclides came from the HCF, which did not begin operation until 1979. This is an assumption favorable to the claimant that will likely overestimate inhalation of these radionuclides.

The estimated inhalation intakes of radionuclides for TA-V are shown in Table 4-6. The material form and solubility should be selected for each radionuclide based on the values most favorable to the claimant.

Table 4-6. SNL-NM TA-V maximum annual median intakes (Bq/yr) via inhalation.<sup>a</sup>

Year	Radionuclide						
	H-3	I-129	I-131	I-133	I-135	Kr-88	Rb-88
1961	5E+02	N/A	N/A	N/A	N/A	N/A	N/A
1962	5E+02	N/A	N/A	N/A	N/A	N/A	N/A
1963	5E+02	N/A	N/A	N/A	N/A	N/A	N/A
1964	5E+02	N/A	N/A	N/A	N/A	N/A	N/A
1965	5E+02	N/A	N/A	N/A	N/A	N/A	N/A
1966	5E+02	N/A	N/A	N/A	N/A	N/A	N/A
1967	5E+02	2E+00	2E+01	3E+02	1E+03	3E+03	2E+03
1968	5E+02	2E+00	2E+01	3E+02	1E+03	3E+03	2E+03
1969	5E+02	2E+00	2E+01	3E+02	1E+03	3E+03	2E+03
1970	5E+02	2E+00	2E+01	3E+02	1E+03	3E+03	2E+03
1971	5E+02	2E+00	2E+01	3E+02	1E+03	3E+03	2E+03
1972	5E+02	2E+00	2E+01	3E+02	1E+03	3E+03	2E+03
1973	5E+02	2E+00	2E+01	3E+02	1E+03	3E+03	2E+03
1974	5E+02	2E+00	2E+01	3E+02	1E+03	3E+03	2E+03
1975	5E+02	2E+00	2E+01	3E+02	1E+03	3E+03	2E+03
1976	5E+02	2E+00	2E+01	3E+02	1E+03	3E+03	2E+03
1977	5E+02	2E+00	2E+01	3E+02	1E+03	3E+03	2E+03
1978	5E+02	2E+00	2E+01	3E+02	1E+03	3E+03	2E+03
1979	7E+04	2E+00	2E+01	3E+02	1E+03	3E+03	2E+03
1980	2E+05	2E+00	2E+01	3E+02	1E+03	3E+03	2E+03
1981	5E+02	2E+00	2E+01	3E+02	1E+03	3E+03	2E+03
1982	5E+02	2E+00	2E+01	3E+02	1E+03	3E+03	2E+03
1983	5E+02	2E+00	2E+01	3E+02	1E+03	3E+03	2E+03
1984	5E+02	2E+00	2E+01	3E+02	1E+03	3E+03	2E+03
1985	5E+02	2E+00	2E+01	3E+02	1E+03	3E+03	2E+03
1986	5E+02	2E+00	2E+01	3E+02	1E+03	3E+03	2E+03
1987	5E+02	2E+00	2E+01	3E+02	1E+03	3E+03	2E+03
1988	5E+02	2E+00	2E+01	3E+02	1E+03	3E+03	2E+03
1989	5E+02	2E+00	2E+01	3E+02	1E+03	3E+03	2E+03
1990	5E+02	2E-01	2E+01	3E+02	1E+03	3E+03	2E+03
1991	5E+02	2E+00	2E+01	3E+02	1E+03	3E+01	4E+01
1992	5E+02	4E+00	2E+00	4E+02	2E+03	3E+03	4E+03
1993	5E+02	2E+00	2E+01	3E+02	1E+03	6E+03	6E+03
1994	5E+02	2E+00	2E+01	3E+02	1E+03	2E+03	3E+02
1995	5E+02	2E+00	2E+01	3E+02	1E+03	2E+03	6E+00
1996	5E+02	2E+00	3E+01	1E+02	2E+01	8E+03	2E+03
1997	5E+02	2E+00	2E+01	3E+02	1E+03	3E+03	2E+03
1998	5E+02	2E+00	2E+01	3E+02	1E+03	3E+03	2E+03
1999	5E+02	2E+00	2E+01	3E+02	1E+03	3E+03	2E+03
2000	5E+02	2E+00	2E+01	3E+02	1E+03	3E+03	2E+03
2001	5E+02	2E+00	2E+01	3E+02	1E+03	3E+03	2E+03
2002	5E+02	2E+00	2E+01	3E+02	1E+03	3E+03	2E+03
2003	5E+02	2E+00	2E+01	3E+02	1E+03	3E+03	2E+03
2004	5E+02	2E+00	2E+01	3E+02	1E+03	3E+03	2E+03

a. Radionuclide solubility class should be selected to be favorable to the claimant.

#### 4.5.2 Sandia National Laboratories – Nevada

The only identified source of potential radionuclide inhalation at TTR is resuspension of radionuclides from the Clean Slate sites. No other areas of the TTR have been determined to be contaminated with site-originated radionuclides.

As noted in Section 4.2.2, the highest areas of contamination at the Clean Slate sites were initially buried and fenced off after the events in 1963 and a second, larger perimeter fence was set 10 yr later. Lower levels of transuranic radionuclide contamination have been identified south of the original sites and away from the principle occupied areas of the range. These levels are due to the prevailing winds at the time the events occurred (Ikenberry 2006a). Soil sampling is conducted each year to assess the potential for dispersion of transuranic radionuclide contamination. No contamination has been detected except for the areas south of the Clean Slate sites. The Clean Slate sites are remote from the Operations Center, and the annual site environmental reports emphasize that test activity is planned to avoid disturbing these areas.

SNL instituted continuous air monitoring at the TTR airport for a 1-yr period from February 22, 1996, to February 25, 1997, to assess potential radiation dose from resuspension of radionuclides from the Clean Slate sites as part of an investigation in relation to the National Emission Standards for Hazardous Air Pollutants (Duncan and Sanchez 1999). The TTR airport was determined to be the location for the highest calculated dose to an onsite maximally exposed individual and so is likely a location favorable to the claimant for estimating exposure of unmonitored SNL-NV workers. The average annual air concentrations were measured as follows:  $^{241}\text{Am}$  at  $4.1 \times 10^{-18} \text{ Ci/m}^3$ ,  $^{238}\text{Pu}$  at  $1.6 \times 10^{-18} \text{ Ci/m}^3$ , and  $^{239/240}\text{Pu}$  at  $9.5 \times 10^{-18} \text{ Ci/m}^3$  (Duncan and Sanchez 1999). Based on 2,000 hr/yr exposure to these concentrations with a light activity breathing rate of  $3.3 \times 10^{-4} \text{ m}^3/\text{s}$ , the activity inhaled annually is  $3.6 \times 10^{-4} \text{ Bq } ^{241}\text{Am}$ ,  $1.4 \times 10^{-4} \text{ Bq } ^{238}\text{Pu}$ , and  $8.4 \times 10^{-5} \text{ Bq } ^{239/240}\text{Pu}$ .

Because of the early efforts to bury soil with high contamination levels near ground zero, the fencing of the Clean Slate contaminated areas, and the operational avoidance of the Clean Site areas, the air concentrations at the TTR airport in 1996 and 1997 are considered an estimate of radionuclide inhalation for unmonitored workers that is favorable to the claimant. The estimated inhalation intakes of radionuclides for TTR are shown in Table 4-7. The material form and solubility should be selected for each radionuclide based on the values most favorable to the claimant.

Table 4-7. SNL-NV TTR maximum annual median intakes (Bq/yr) via inhalation.<sup>a</sup>

Year	Radionuclide		
	Am-241	Pu-238	Pu-239/240
1957–1962	0	0	0
1963–2004 <sup>b</sup>	5 E-04	2 E-04	1 E-04

- Radionuclide solubility class should be selected to be favorable to the claimant.
- Potential inhalation after 1963 is estimated on dispersion of contaminants from the Clean Slate sites (see Section 4.2.2)

#### 4.6 **INGESTION**

##### 4.6.1 Sandia National Laboratories – New Mexico

There is no evidence of the potential for inadvertent ingestion intakes at SNL-NM. Therefore, no estimates of ingestion intakes are made and ingestion intakes should be considered zero.

#### **4.6.2 Sandia National Laboratories – Nevada**

There is no evidence of the potential for inadvertent ingestion intakes at SNL-NV. Therefore, no estimates of ingestion intakes are made and ingestion intakes should be considered zero.

#### **4.7 UNCERTAINTY**

Uncertainty is lower in more recent years and higher in earlier years when less data are available. There are no documents available from which quantitative estimates of uncertainty can be made, so qualitative estimates are provided.

##### **External Dose**

A qualitative estimate of SNL-NM occupational environmental external dose uncertainty is based on a normal distribution with a standard deviation of 30 percent for all exposure periods and TAs. This is applicable only for unmonitored workers.

The estimate of occupational environmental dose to SNL-NV workers at TTR is estimated to be zero, so no uncertainty is provided.

##### **Inhalation Intakes**

There are no available documents with estimates of uncertainty for SNL-NM stack releases. However, the potential for environmental inhalation of radionuclides appears to be lower during the early years of SNL-NM activities (in contrast to many production sites during the Cold War era). A qualitative estimate of inhalation intakes assumes lognormally-distributed chronic intakes, with uncertainty of a geometric standard deviation (GSD) of 2 for all exposure periods.

A qualitative estimate of inhalation intake uncertainty at SNL-NV is also based on a lognormal distribution with GSD of 2 for all exposure periods.

##### **Ingestion Intakes**

The estimate of occupational environmental ingestion intakes by both SNL-NM and SNL-NV workers is zero, so no uncertainty is provided.

#### **5.0 OCCUPATIONAL INTERNAL DOSE**

Occupational internal dose is the dose received by an individual from an intake of radioactive material while performing tasks within buildings and structures at SNL (which was operated as the Z-Division and later Albuquerque Branch of LASL from 1945 until 1949) or from activities outside the buildings, such as burial of waste and monitoring of tests, where occupational intakes of radioactive material could occur. This document contains information for reconstruction of occupational internal doses at SNL facilities throughout its history.

#### **5.1 INTRODUCTION**

Operations began at SNL (then Z-Division of LASL, Albuquerque Branch) in July 1945. SNL was originally created to perform ordnance engineering and assembly aspects of LASL's design work. In essence, Z-Division was responsible for all the nonnuclear components of nuclear weapons, whether through internal assembly or procurement. In late 1945, the LASL began transferring its field testing and engineering organization, known as Z-Division, to Sandia Base near Albuquerque. Staff from the Army Air Corps 509th Composite Group at Wendover Air Base in Utah joined the original group to assemble weapons. This organization formed the nucleus of Sandia Laboratory, created in 1948 as a separate branch of Los Alamos.

The following year, the laboratory formally separated from Los Alamos when the University of California, Los Alamos' managing contractor, asked to be relieved of the responsibility. American Telephone and Telegraph (AT&T), at the request of President Truman, agreed to take over management of the facility, and Sandia Corporation, a wholly owned subsidiary of Western Electric and AT&T's production arm, was formed to serve as the managing contractor [5].

By 1952, the weapons production complex was in place. SNL focused on weapons development and expanded its engineering staff to accommodate the growing number of weapons projects underway. In addition to design and production coordination, SNL undertook extensive field testing of components and supported the atmospheric tests sponsored by its partner laboratories. The bulk of the work at SNL and its properties has been related to the nonnuclear aspects of nuclear weapons design. This work includes weapons design and testing, production engineering, stockpile maintenance, and stockpile surveillance (Ullrich 1998).

From 1945 to the present, SNL employees have routinely been involved in operations at a number of Sandia Corporation sites. These sites include Livermore, California; Hattiesburg, Mississippi; NTS, Nevada; Clarksville, Tennessee; and Salton Bay Station, California. SNL employees also routinely spent time at other DOE facilities. Dosimetry monitoring records from these offsite activities might or might not be available in the SNL dosimetry records [6]. SNL has also been responsible for surveillance of the nation's stockpiled weapons (Ullrich 1998). Since 1949 when weapons storage sites were opened and until 1967, SNL stationed staff at the storage sites to monitor, maintain, and assemble the weapons [7].

SNL operates major research nuclear reactors and electron/ion accelerators for the DOE Office of Military Applications. Research and development activities are conducted in relation to nuclear weapons systems, nonnuclear weapons systems, advanced nuclear reactors, simulation source development, and other basic and applied research areas at each facility [8].

The potential for chronic intakes at SNL is far less than at DOE production sites because of the nature of the tasks performed at SNL. Certain areas of the site are nonnuclear. In addition, the nature of the research environment at the SNL site results in intake potentials that are often unique and of short duration. Nevertheless, the potential for monitored and unmonitored intakes has existed throughout the history of the site [9].

### **Assembly**

*Assembly* is used in a variety of contexts and conveys a variety of meanings, especially in the early history of SNL (Ullrich 1999). The resultant confusion can make it difficult to determine what the exposure potential of a worker might be, given statements of "assembly" of weapons in either the telephone interview or the other work histories [10].

Although the Little Boy gun-type nuclear weapon received some research and development attention after World War II, the majority of the early postwar nuclear weapons were implosion weapons based on the Manhattan Project's Fat Man design. In this design, a sphere of high explosive surrounded a central core of nuclear material. Detonation of the high explosive compressed the central core of nuclear material to a supercritical mass. Introduction of neutrons to the highly compressed core initiated the nuclear chain reaction. However, the nonnuclear components including the large high-explosive lenses were assembled and stored separately from the nuclear material and components that turned them into nuclear weapons. To use a weapon, it had to be partially disassembled, the nuclear core inserted, and reassembled. Sealed-pit weapons with their nuclear components installed during assembly did not enter the stockpile until 1957.

The nature of these weapons meant that individuals using the word *assembly* during the 1940s and 1950s might mean any of a number of things. Assembly could mean the final assembly before a weapon was used or tested (that is, inserting the nuclear core into the weapon). An alternative meaning (most frequently in the case of Z-Division activities) for assembly was to put all of the nonnuclear pieces, including the high-explosive lenses, together for storage or testing. The latter use also segued into the description of putting together inert mock-ups or prototypes of weapons (that is, assembling a weapon with neither high-explosive nor nuclear core). Last, assembly was also used to refer to putting appropriate pieces together into components, leading up to final assembly of one of the other sorts.

Certain buildings, such as Building 828, were not used for either high-explosive lens assembly or nuclear core insertion. Other facilities were available and used for that purpose. All evidence indicates that Building 828 was designed and used as a mechanical test laboratory. As such, it represents the core of SNL's early ordnance design mission (Ulrich 1999).

Therefore, terms similar to *weapons assembly* in the telephone interview or work history do not necessarily indicate potential exposure to radionuclides.

### Nuclides

Nuclides with the widest historical and current application throughout the SNL facilities are:

- Tritium ( $^3\text{H}$ )
- Uranium ( $^{238}\text{U}$ ,  $^{234}\text{U}$ ,  $^{235}\text{U}$ )
- Fission and activation products (e.g.,  $^{90}\text{Sr}$ ,  $^{137}\text{Cs}$ ,  $^{65}\text{Zn}$ ,  $^{60}\text{Co}$ ,  $^{182}\text{Ta}$ )

And to a lesser degree,

- Plutonium ( $^{238}\text{Pu}$ ,  $^{239}\text{Pu}$ ,  $^{240}\text{Pu}$ ,  $^{241}\text{Pu}$ )
- Americium ( $^{241}\text{Am}$ ) [11].

These radionuclides of primary concern are listed in Laboratory reports from 1945 to the present. Potential exposures to nuclides were encountered in Technical Areas (TA) and test facilities (Potter 1998a) as listed in Table 5-1. More detailed descriptions of the TAs and potential hazards are found in Section 2.0. If the specific work areas are known, the dose reconstructor should review Section 2.0, Table 5-1 and Table 5-2 to determine the potential for exposure to internal hazards. Many of the buildings did not have a potential for routine intakes of radioactive material or were completely devoted to nonradioactive functions. Table 5-2 lists the significant buildings and areas in TA-III and TA-IV with the potential for worker exposure to external radiation or radioactive materials.

Table 5-1. Internal exposure potential by area (Potter 1998a) [12].

TA (start year)	Uranium	Tritium <sup>c</sup>	Plutonium	Americium <sup>b</sup>	Other (Sr-90, Th, tracers, etc.)
TA-I <sup>f</sup>	DU Bldg 805-807, Bldg 869 Toxic Shop, machining, 840 after closure of 869 c1985, 849, 819, 835, 892 and 809 (bare DU pieces stored on cabinet)	Bldg 891 as erbium tritide; <sup>d</sup> Bldg 884, 807, 844, 819	Sealed Pu-238 heat sources, 819 (nuclear materials receiving)	Environmental restoration sites	Environmental releases of accelerator activation products (O-15, N-13, Ar-41), 869 Toxic Shop, machining of activated stainless steel and fission products (VanDevender 1984), FP 805, IA, 805, 807; 818 sealed sources.
TA-II <sup>e</sup> (1948) through 1960s	906 Decon Bldg 9939, U melt facility, waste, high	HT or HTO			906 FP, Decon Bldg

TA (start year)	Uranium	Tritium <sup>c</sup>	Plutonium	Americium <sup>b</sup>	Other (Sr-90, Th, tracers, etc.)
	contamination				
TA-III	DU, Entire area <sup>a</sup>	Entire area <sup>a</sup>	Pu contaminated wastes. <sup>j</sup>		RMWMF, FP, Co-60, thorium <sup>j</sup>
TA-IV <sup>k</sup>		PBFA-II and SABER accelerator targets <sup>i</sup>			PBFA-II Fe-59, Fe-55 activation products, aerosol when machined.
TA-V	ACRR, HCF, <sup>g</sup> SPR	ACRR, HCF, <sup>g</sup> SPR	ACRR, HCF, <sup>g</sup> SPR	ACRR, HCF, <sup>g</sup> SPR	Fission products from reactors
Coyote Test Field (CTF)	DU, Entire area <sup>a</sup>				
TTR	DU, Entire area <sup>a,h</sup>		Entire area <sup>a,h</sup>	Entire area <sup>a,b</sup>	
Manzano	Waste storage facility				

- a. Also at environmental restoration sites and RMWMF.
- b. As a portion of the plutonium mixture and as pure Am-241 (Stanley 1993).
- c. Some facilities use tritiated solvents that might require special biokinetics.
- d. Research suggests that the retention time of metal tritides (MTs) is longer than that of HTO (Inkret et al. 1999).
- e. 1948 to 1952 primary site for weapons assembly (subassembly level, no nuclear pits in the weapon). Facilities for small explosive component research and testing late 1950s to early 1960s. After 1960 the primary purpose of TA-II was the research and testing of high-explosive components. Buildings 904 and 907 are the building of major significance in TA-II (Potter 1998a). Tritium contained in test components, although it is unclear if this was the condition during the entire history of the site.
- f. Nonnuclear components were assembled by the Road group in TA-I for shipment to TA-II for completion of the subassembly.
- g. Area consists of the hot cell laboratory, the glovebox laboratory (10 high-purity gloveboxes) and the analytical laboratory. Hot cell laboratories consist of three steel containment boxes for the assembly and disassembly of experiments that can contain 6,000 Ci of fission products and/or 500 Ci of plutonium. The gloveboxes can contain 300 Ci of fission products.
- h. No bioassay is routinely done for uranium at TTR because workers are not expected to encounter 2 µCi of plutonium or 10 µCi of DU. Baselines might have been performed (Potter 1993a).
- i. An incident involving a leaking neutron calibration device in April 1991 also produced tritium intakes (Burnett 1991).
- j. From Section 2.0.
- k. See Table 5-2 for details of specific areas in TA-IV.

Table 5-2. Facilities within TA-III, TA-IV, and the Explosive Component Facility (ECF).

Technical area	Building	Facility	Radiation	Remarks
TA-III		Thermal Treatment Facility (SNL 1998b)	No radioactive material is treated	
		MWL (SNL 1998c)	Low-level waste (LLW), mixed waste	1957-1988
	6630		Melt furnace contaminated with uranium (Tucker 1977)	
	6930		Reactor experiment, activated materials (Tucker 1977)	
	6921		Building with uranium stored (Tucker 1977)	
	6920, 6921, 6925	RMWMF (SNL 1998c)	LLW, transuranic waste, mixed waste Sealed sources stored or in stack monitors.	1988 to present treatment and packaging
TA-IV	Simulation Technology Laboratory/ Building 970	HERMES III (high-energy, linear induction accelerator (SNL 1998d)	External and O-15 and N-13 only, short-lived activation products (Zn-65, Co-57, Na-22, Mn-54)	40 weeks/yr, 9 weeks operational set up and maintenance
		SABRE (pulsed accelerator) (SNL 1998e)	External and activation products require a 30-min delay before entry and handling, maximum a few days for reuse. Sealed source Cm-244, Na-22	Items stored for decay before cleaning and reuse.
		PROTO II (accelerator)		

Technical area	Building	Facility	Radiation	Remarks
	981	SATURN (Pulsed Power Technology) (SNL 1998f)	Some activation of Be-Cu materials and external.	
	981	SPHINX accelerator (SNL 1998g)	External, no activation products are made.	
	961	TESLA accelerator (SNL 1998h)	external	
	963	Advanced Pulsed Power Research Module (APRM) (SNL 1998i)	external	
	960	(Jow 1991)	Ni-63 (<10 µCi each), Am-241 calibration sources,	
	962	(Jow 1991)	Mixed activation materials and parts from NTS, storage for units containing <100 µCi Kr-85, <10 µCi Ni-63, and 1,000 µCi H-3.	
	983	Z Machine (formerly PBFA II – PBFA II) (SNL 1998j)	Tritium; low level of water concentration. Nuclear targets; tritium, Pu-239, DU Activated hardware, 50,000 kg	Neutron shots 200 mg Pu and DU 1,000 Ci tritium
	959	Hazardous Waste Management Facility (HWMF) (SNL 1998k)	No radioactive materials	
		Particle Beam Fusion Accelerator (PBFA II) (VanDevender 1984)	Activated stainless steel	Aerosols generated during machining
Outside secured area	897	Integrated Materials Research Laboratory (IMRL) (SNL 1998l)	DU, C-14 Sealed sources	Millicurie quantities
ECF	950	ECF (SNL 1998m)	10 µCi “Barium bolts” (Ba-133) and 100 mCi gaseous tritium and MT.	

Many of the exposure histories and work records are not specific about the work areas to which individuals were assigned. However, when information about the work location is available, Table 5-1 can be used to determine the probable nuclides. In addition, workers were involved in significant offsite activities involving weapons testing or other related activities. Table 5-3 lists some of these tests to assist the dose reconstructor in identifying terms that might be referenced in personnel records or the telephone interview [13]. Workers whose routine jobs did not involve work with nuclear materials could have encountered internal exposures while participating in weapons testing. The dose reconstructor is referred to the TBDs for the specific sites for minimum detectable activities (MDAs) for these bioassay results when MDAs are not listed with the results.

Table 5-3. Nuclear weapons and nosetip<sup>a</sup> tests (not all-inclusive) (SNL 1992b).

Test name	Date	Details
Crossroads	Summer 1946	Two test shots in the Bikini Atolls to assess the effects of nuclear weapons on ships at sea. Shot Able was an air drop. Shot Baker was an underwater detonation.
Sandstone	1948	
Hard Hat	02/16/1962	NTS, Seepage through stemming
Marshmallow	06/28/1962	NTS, Massive stemming failure, release continued for several days
Shoal	10/26/1963	Fallon, NV, release of 110 Ci of Xe-131m, <1 Ci I-131.
Gumdrop	04/21/1965	NTS, controlled ventilation, primarily Xe-135
Diluted Waters	06/16/1965	NTS, massive stemming failure, gross fission products released
Tiny Tot	06/17/1965	NTS, seepage through shaft, Xe-138, Xe-135, Kr-87, Kr-88, 20 Ci

Test name	Date	Details
		of radioiodines.
Red Hot	03/05/1966	NTS, release of noble gasses, I-135 (2000 Ci), I-133 (500 Ci), I-131 (20 Ci)
Pinstripe	04/25/1966	NTS, gross fission product release
Piledriver	06/02/1966	NTS, only Xe-135 detected in release
Double Play	06/15/1966	NTS, primarily noble gasses released
Derringer	09/12/1966	NTS, noble gasses, I-135 (152 Ci), I-133 (41 Ci), I-131 (1.5 Ci)
Newpoint	12/13/1966	NTS, primarily noble gasses
Midi Mist	06/26/1967	NTS, noble gasses and radioiodines in release
Door Mist	08/31/1967	NTS, noble gasses with radioiodines (I-135, I-133, I-131) and ruthenium (Ru-103 and Ru/Rh-106), primarily
Dorsal Fin	02/29/1968	NTS, no release
Milkshake	03/25/1968	NTS, primarily Xe-138
Diana Moon	08/27/1968	NTS, seepage, Xe-138 and radioiodines (I-135, I-133, I-131); later release of Xe-135 and (I-135 [3.6 Ci], I-133 [2.1 Ci], I-131 [0.1 Ci])
Hudson Seal	09/24/1968	NTS, No release
Ming Vase	11/20/1968	NTS, no release
Cypress	02/12/1969	NTS, no release
Minute Steak	09/12/1969	NTS, release of Xe-138, Xe-135, Xe-133 and (I-135 [34.4 Ci], I-133 [3.4 Ci], I-131 [0.05 Ci])
Diesel Train	12/08/1969	NTS, no release
Diana Mist	02/11/1970	NTS, no release
Mint Leaf	05/05/1970	NTS, release initially primarily of Xe-135, later of Xe-133m and Xe-133 and some radioiodines.
Diamond Duet	05/12/1970	NTS, release of fission gasses, Xe-133m and Xe-133
Hudson Moon	05/26/1970	NTS, release of Xe-135, Kr-88 and Kr-85m
Diamond Mine	07/01/1971	NTS, no release
Grommet	07/1971–06/1972	NTS
Diagonal Line	11/24/1971	NTS, primarily Xe-135, lesser amounts of Kr-85m, Kr-87, Kr-88, Xe-131m, Xe-132, Xe-133, Xe-133m and trace I-131, I-132, I-133 and I-135.
Misty North	05/02/1972	NTS, No release
Toggle	07/1972–06/1973	NTS
Diamond Sculls	07/20/1972	NTS, No release
Dido Queen	08/05/1973	NTS, No release
Husky Ace	10/12/1973	NTS, No release
Ming Blade	06/19/1974	NTS, No release
Payload–ANT-I	1974–1977	Vandenberg, nosetip contained Sr-90/ Se-75/ Ta-182, external surface contamination of Ta-182 (SNL 1977b)
Hybla Fair	10/28/1974–01/06/1975	NTS, Xe-133 and Xe-133m
Dining Car	04/05/1975	NTS, No release
Husky Pup	10/24/1975	NTS, No release
Mighty Epic	05/12/1976	NTS, No release
Payload SAMAST 3/ MINT 2	1976	Vandenberg, Se-75, DU (ballast) and Co-57 sealed sources. Ballast installed at Union Carbide.
Hybla Gold	11/01/1977	NTS, No release
Payload–ANT-II and ANT-III	1977–1980	Vandenberg, potential for release of Ta-182, nosetip, external surface contamination of Ta-182 (SNL 1977a)
Diablo Hawk	09/13/1978	NTS, No release
Huron King	06/24/1980	NTS, No release
Miners Iron	10/31/1980	NTS, No release
Huron Landing	09/23/1982	NTS, Xe-133, Xe-133m, Xe-135, Kr-85m, Kr-88

Test name	Date	Details
Mini Jade	05/26/1983	NTS, Xe-133 and Xe-133m
Tomme/ Midnight Zephyr	09/21/1983	NTS, No release
Midas Myth	02/15/1984	NTS, No release
Midas Rain	04/06/1985	NTS, Xe-133, Xe-133m, Xe-135
Mill Yard	10/09/1985	NTS, Xe-133, Xe-133m, Xe-135
Diamond Beach	10/09/1985	NTS, Xe-133, Xe-133m, Xe-135
Mighty Oak	04/10/1986	NTS, Xe-133 and 2.4 Ci I-131
Middle Note	06/20/1987	NTS, No release
Mission Ghost	06/20/1987	NTS, Kr-85
Mission Cyber	12/02/1987	NTS, No release
Misty Echo	12/10/1988	NTS, No release
Disko Elm	09/14/1989	NTS, Xe-133, Xe-133m, Xe-135
Mineral Quarry	07/25/1990	NTS, No release
Distant Zenith	09/19/1991	NTS, No release
Diamond Fortune	04/20/1992	NTS, Xe-133 and traces of I-131
Hunters Trophy	09/19/1992	NTS, No release

a. Nosetip tests involve exposure to potential intakes of Ta-182 and, possibly, Se-75 only as described above.

The SNL mission was primarily that of research, mostly focused on the effects of environmental factors and external radiation exposure on weapons materials and components. No potential for internal exposure was anticipated in many of the areas on the site [14].

There is very little documentation about the internal dosimetry program before 1992. However, several references to compliance with the AEC requirements for limiting dose have been found. Although these documents were published in the 1960s and early 1970s, it appears that the AEC requirements for limiting dose to 0.25 rem/year (1/20<sup>th</sup> of the Maximum Permissible Annual Dose [MPAD]) extends to earlier years where AEC compliance was also required (Lane 1957). Formal information about bioassay techniques or frequency also is not readily available before 1992. The Industrial Hygiene department was responsible for bioassay for uranium from at least 1959 through 1991 (Potter Sturgis 1993), and possibly as early as November 1949 (Argall 2007a, b). Tritium and possibly some uranium and plutonium, bioassay were also performed [15]. Less than 100 workers participated in the bioassay program each year from the beginning of the program through 1991. The records of the bioassay were transferred to the corporate archives as microfilm or microfiche and dose summary results were entered into an electronic database beginning in 1967 (Argall 2007b). Much of the *in vivo* bioassay was out-sourced until the SNL whole body counter became operational in 1993.

In 1991, a DOE Tiger Team assessment team identified several issues with the internal dosimetry program at SNL (SNL 1995a, 1995b). The details of these issues are addressed later in the document.

The results of bioassay samples analyzed between August 1992 and April 1994 by a commercial laboratory were invalidated based on the results of spike samples and other issues surrounding the integrity of the data (Potter 1994a; DOE 1994) [16]. Records were found for uranium lung counting performed for workers in 1989 and 1990 by Helgeson. A positive bias, based on results of unexposed workers, is indicated for these counts (Brake 1989) [17].

Tritium, gamma isotopic, and uranium bioassays are currently being performed on the site. Whole-body counting is also being performed on the site. Work areas include the plutonium areas (<sup>239</sup>Pu and

heat source ( $^{238}\text{Pu}$ ), uranium areas, tritium facilities, laboratory facilities, reactors, accelerators, and environmental restoration sites.

Many SNL workers received exposure and intake monitoring at sites other than SNL locations. Those attending the nuclear tests and, to a lesser degree, nosetip tests (as described in Section 5.1 in this document) may have had a potential for intakes. These tests were given names. The names and dates of selected nuclear weapons tests (shots) and nosetip tests are listed in Table 5-3. Workers whose routine jobs did not involve work with nuclear materials could have encountered potential internal exposures while participating in weapons testing. Work at off site locations is indicated by either the Department of Energy records or the telephone interview [18]. Offsite dosimetry records are typically not contained in the SNL records, but are supplied directly from the appropriate sites [19]. If the telephone interview or work history indicates that the worker participated in nuclear weapons testing, and if no dosimetry records are found for that period, then the individual site profiles can be reviewed to obtain the information necessary to calculate potential missed dose. While all tunnel entries involved potential exposure, some of the tests also had controlled or uncontrolled effluent releases. These releases are noted in Table 5-3.

Nosetip tests had an intake potential different from that of weapons tests. The intake potential would be to a limited number of nuclides ( $^{182}\text{Ta}$  is the most significant). Nosetips containing TaC rods were irradiated to produce  $^{182}\text{Ta}$ . Some instances of surface contamination caused by the irradiation of materials on the nosetip or heat shield have occurred during tests. The most significant hazard in these tests was external radiation although there was some potential for intakes from surface contamination caused by tramp elements on surfaces before irradiation. The nosetips in these tests were typically prepared at SNL and shipped to Vandenberg Air Force Base. A health physicist from SNL would travel to Vandenberg to perform the surveys on the shipment when it arrived and during the time the shipment was prepared for the test. The health physicist would take contamination surveys and external dose readings (SNL 1977a).

Section 5.2 discusses *in vitro* methods for specific radionuclides. Because the perception was that the potential for intakes at SNL was small, the routine bioassay program was not formalized until after 1992. Before that time, monitoring for intakes was not regularly performed. Some bioassay for tritium and uranium was performed beginning in 1949 [20]. A memorandum of understanding (MOU) was in place with LANL for bioassay (Potter 1994a; Ball et al. 1995); whole-body counts and lung counts that might have been required. Records indicate that Helgeson Scientific Services was contracted to perform whole-body counts and lung counts for uranium once in 1989 and possibly again in 1990 (HSS 1989). Air sampling occurred infrequently, although health physics technician's logs mention continuous air monitor (CAM) alarms and malfunctions. Records indicate that CAMs were located in the Glovebox Laboratory (GBL) of the SER facility from at least 1957 (Lane 1957). In the early years only workers with a significant potential for exposure were monitored. Although the number of individuals monitored has increased, not all individuals working at SNL are currently monitored. Bioassay occurred either as confirmatory measurements for air-sampling results, or air sampling occurred as confirmatory for positive bioassay results. Some air-sampling limits were found from as early as 1964. Recent data capture efforts at SNL have identified large volumes of bioassay data. These data will be used in the coworker analysis. Once this analysis is completed the TBD will be revised to incorporate the use of these data for claims that warrant an internal dose assessment but lack monitoring data in their case files provided by DOE.

Section 5.3 discusses *in vivo* bioassay methods used currently and historically at SNL.

### 5.1.1 Bioassay Results of Individuals

There are few results of bioassay of individual SNL workers found in the electronic records prior to 1989. An interview with a retiree who was involved in the archiving of early dosimetry records indicated that records of bioassay beginning in 1949 were entered into an electronic database when the bioassay results were being microfilmed [21]. Of the claimant records reviewed, most of the bioassay results in the records are from temporary assignments at sites other than SNL, although the person may have actually been assigned to SNL at the time that the work at another DOE or SNL site was performed. Bioassay records before 1989 have been found in paper and microfiche format in the SNL archives, retrieval is labor intensive. These records contain uranium, plutonium, and tritium bioassay results. Searches are being done as the claims are received and to support coworker dose analysis. Also, it is likely the bioassay records and external dosimetry records for workers employed by SNL prior to November 1, 1949 [22], when the Z-Division officially separated from LANL, may only be located in the LANL Bioassay database. If the employment dates prior to November 1, 1949 are listed for a claimant, then dosimetry records should be requested from LANL if none are supplied in the energy worker's file.

A 1989 appraisal of Health Physics and Industrial Hygiene indicates that there were no procedures for the Internal Dosimetry Program detailing employees on bioassay, frequency of sampling and responsibility for results (Hyde 1989). The report indicated that there was good radiochemistry support to the bioassay analysis using documented procedures for nonquantitative uranium and possibly tritium (SNL 2005a, b, c, d).

Brake (1989) suggests that the large number of positive  $^{238}\text{U}$  and  $^{235}\text{U}$  positive lung counts reported by Helgeson may have been attributed to the acknowledged positive bias exhibited by the Helgeson system. Also, bioassay samples were sent to Controls for Environmental Pollution (CEP, an offsite commercial laboratory) from August 1992 to April 1994. However, poor performance on blank and spike samples were noted early in the contract. An investigation in 1995 subsequently invalidated all of these results (Ball et al. 1995). Forty-two cases in which there was a potential incident were scheduled for follow-up. Ten of the 492 baseline or routine bioassays were resampled. However, because the follow-up occurred as much as 2 yr after the potential exposures, the results for analytes such as tritium and fission products would not have been representative of potential intakes [16]. Some bioassay analyses for tritium were performed by the SNL Industrial Hygiene or NTS during that period. Those results are considered valid. Before the contract was awarded to CEP, samples were out-sourced to Atlan-Tech. This contract was dropped in favor of CEP because of poor performance of Atlan-Tech. Therefore, results from Atlan-Tech should also be considered suspect (Ball et al. 1995).

Tritium results for SNL personnel on assignment at LLNL could be reported in whole-body dose rather than as *in vitro* bioassay results [22]. The bioassay results from LLNL are available for the dose reconstruction upon request. Assignment of tritium dose for these individuals should follow the protocol established for workers permanently assigned to LLNL. This protocol for tritium dose is discussed in Section 5.2.3 and in *Lawrence Livermore National Laboratory – Occupational Internal Dose* (ORAUT 2005b).

There is evidence that some workers may have been monitored by the Industrial Hygiene group using breathing-zone air (BZA) samples (Pigg 1993). However, it is not known if the results of the BZA samples are included with the individual dosimetry records provided by DOE.

### 5.1.2 The Bioassay Program

Before 1992, the program was directed by the individual groups and areas. Only one full-time employee was assigned to the internal dosimetry program. The Industrial Hygiene group was “in charge” of the bioassay (Potter and Sturgis 1993). The respiratory protection program was administered by the Industrial Hygiene group. Individuals were assigned to monitoring following suspected intakes prompted by incidents, high results of general air samples, CAM alarms, or nasal swipes (Hasenkamp 1961). Administrative limits were based on the prevailing AEC or DOE guidance (e.g., ICRP Publications 2 [ICRP 1959] and 30 [ICRP 1979]). Discussions with a retiree indicate that bioassays were performed for uranium, plutonium, and tritium as early as 1949 (Argall 2007a, b). Bioassay samples may have been sent off the site for analysis. As discussed previously, the assumption was that the potential for intakes was low (Gonzales 1985) and that the particle sizes of radioactive materials dispersed in the shots were generally accepted as larger than respirable as described in a conference call with a retiree (McConn 2006), and discussed in a uranium briefing (Jow 1993).

The Final Headquarters Report on the Nuclear Safety Program Appraisal of the DOE Albuquerque Operations Office in June 1985 described the internal dosimetry program as follows (Gonzales 1985):

*The SNLA has minimal need for an internal Dosimetry program and thus the program is virtually nonexistent. No whole body or lung counting is routinely being done, however urine sampling is scheduled for a few employees. The following findings are provided:*

*Finding #15: The most recent Internal Dosimetry procedure available was dated 1977. Some substantive changes had since been written in but the procedure had not been updated.*

*Finding #16: The quarterly collection of urine samples which are processed for tritium is ineffective. If workers are to be routinely monitored for possible tritium exposure most facilities would collect samples for tritium analysis at least weekly.*

*Finding #17: The quality assurance audit program is not documented and extremely limited in scope.*

Routine sampling was not a general practice. Bioassay was performed in response to suspected intakes of radioactive materials or whether the worker would exceed an administrative threshold (Hasenkamp 1961). Baseline bioassay was required of individuals with a history of exposure to radioactive material. Termination bioassay was performed when an individual no longer qualified to participate in the program because of changes in work conditions or termination of employment. Management practice was to determine the need for participation in a bioassay program on the likelihood of exceeding dose guidelines. A 1992 internal memorandum regarding the participation of the staff of Dept. 6521 in the bioassay program suggests that the staff be tested once after a scheduled maintenance and the next operation involving high surface contamination. The criteria for participating in further bioassay would be “based upon the likelihood of ingesting a 100 mR/yr burden, staff desire to be tested, results from the above mentioned one-time tests, knowledge of program costs, DOE requirements, and health physics advice” (Bryson 1992).

Security inspectors were required to participate in a routine bioassay program and have annual whole-body counts and urinalyses. If routine bioassay of radiation workers who performed hands-on work indicated positive exposures, required bioassays would be expanded to include security

inspectors who could have been present when the exposures occurred. Baseline and termination bioassay were required (SNL 1992c).

The SNL bioassay program was formalized beginning in 1992 in response to an assessment by the DOE Tiger Team. An assessment of the program in 1995 by Kenneth Skrable (Skrable 1996) further clarified aspects of the program that required refinement. This resulted in a revision to the *Technical Basis Document for the Internal Dosimetry* program in 1998 (Potter 1998a). The requirements of 10 C.F.R. pt. 835, "Occupational Radiation Protection," were incorporated into the program.

### 1991 Tiger Team Assessment Findings

Table 5-4 lists key findings and action item responses from the 1991 assessment (SNL 1995a, 1995b). These findings indicate that the internal dosimetry program was not extensive or centralized during the years immediately preceding the assessment. Aspects of the program existed but might not have been implemented site-wide, or for other reasons, were not being consistently implemented. While this assessment only captures the conditions that existed in 1991 (during the transition of the program to the new DOE orders and 5480.11 [DOE 1988]), no records have been located that indicate that the program was more extensive in earlier years although health physics concerns were being addressed in individual areas. An example is a memo of record dated April 27, 1965, which details a trip to investigate the health physics problems related to TRIGA reactors as related to the health physics operations of the Sandia Pulsed Annular Core Reactor (Tucker and Tucker 1965). Several of the findings relate to the delay in implementing changes specific to the new system; others are more generic. The DOE Headquarters' report discussed above and the Tiger Team findings listed below provide a summary of the program and possible shortcomings.

Table 5-4. 1991 Tiger Team assessment key findings related to the internal dosimetry program.

Finding no.	Description	Action plan response	Closure date
16	KF-SH-01; lack of compliance with DOE Orders, for Radiation Protection 14 of 24 areas were in noncompliance.	Establish a program and staffing to come into compliance.	05/94
379	Radiation protection activities are being performed without definitive written guidance or rigorous professional oversight, resulting in inconsistency and noncompliance with DOE requirements.	A series of procedures was available by 6/91. Others were scheduled to be modified.	06/98
387	A formal permit system for control of radiation work has not been implemented.	5 new radiation protection programs developed. The programs include procedures specifying protective clothing requirements, respiratory protection requirements, completing radiological work permits (RWPs), HP document control, use and approval of RWPs and engineering design in the workplace.	06/93
393	Personnel are permitted to enter and work in areas with a potential for airborne radioactive contaminants that have not been monitored, as required by DOE 5480.11 (DOE 1988).	A new program for monitoring workplace air was developed. Equipment was to be installed in 1996.	01/96
397	Glovebox gloves are frequently patched rather than replaced as industry practice dictates.	Modified procedure to survey for contamination and replace gloves if damaged.	09/92
400	ALARA program does not fully meet the requirements of DOE 5480.11 (DOE 1988) and other DOE guidance.	Procedure was being rewritten.	12/94
401	Personnel exposure files do not contain	The files will be expanded to include relevant	04/96

	all the radiological information related to personal exposures and radiological working conditions as required by DOE 5480.11 (1988).	data.	
402	Internal Dosimetry records are not generated and maintained as required by ANSI N13.6 (ANSI 1972) and DOE 5480.11 (DOE 1988).	The Personnel Internal Radiation Dosimetry program was under development. A commitment was made to comply. [Noncompliance with ANSI 13.6 (ANSI 1972) was cited in an assessment in 1985 (Gonzalez 1985).]	08/96
405	Management has not implemented a system to assure ES&H Department review, approval, and control of all radiation protection activities.	Implement process to ensure ES&H review of activities and RWPs.	12/91
452	Respiratory Protection Program does not comply with ANSI Z88.2-1980 (ANSI 1980), DOE 5480.11 (DOE 1988) and 29 CFR 1910.134.	Approved new program and began quantitative testing for all negative pressure respirators in 4/1991. However full implementation was not scheduled unit 1995.	10/95
535	Surveillance of work area conditions during respirator use was not adequate. Also, sufficient documentation to support respirator selection and use did not exist.	All planned items to be addressed.	08/95
536	The breathing air garment developed for the Tritium Research Laboratory had not been approved by the DOE.	Ceased use until approval of the garment could be completed.	08/90

### Current Internal Dosimetry Program

SNL relies on engineering controls to prevent intakes. Bioassay is used as a confirmation that the engineering controls are functioning properly. An example is the annual whole-body counts received by radiological control technicians and waste handlers. Waste handlers also receive routine bioassay for tritium. No other routine programs are in place [23].

Beginning in 1994, workers were required to participate in the bioassay program related to their entry into posted areas. Table 5-5 lists the current requirements. Table 5-6 lists historical requirements as found in the records. Baseline bioassay is not specifically defined, but is assumed to include either the full set of bioassay and whole-body counts or the bioassay related to the area and type of work to be performed. Potter (1998b) describes baseline sampling as "bioassay procedures for new workers who may be exposed at SNL to significant technologically enhanced levels of naturally occurring radionuclides or to synthetic radionuclides to which they were exposed prior to work at SNL." Visitors were not required to submit baseline samples.

Table 5-5. Internal dosimetry requirements (current) (Potter 1994b).

Posted Area	Internal dosimetry requirements <sup>a</sup>
Entry into Controlled Area or Radioactive Materials Area	None
Entry into Radiation Area	Baseline bioassay (radiation workers only)
Entry into High Radiation Area	Baseline bioassay (radiation workers only)
Entry into Very High Radiation Area	Baseline bioassay (radiation workers only)
Entry into Contamination Area	Review by RPID <sup>b</sup> to establish bioassay
Entry into High Contamination Area	Review by RPID to establish bioassay
Entry into Soil Contamination Area	Review by RPID to establish bioassay
Entry into Airborne Radioactivity Area	Review by RPID to establish bioassay
Entry into Tritium Contamination Area	Review by RPID to establish bioassay

- a. In addition, individuals wearing respiratory protection for radiological purposes, or who perform contamination and/or airborne surveys, and emergency response personnel shall participate in the internal dosimetry requirements. Termination bioassay was required of anyone who participated in the bioassay program. The program also included routine and job-specific sampling requirements.
- b. RPID = Radiation Protection Internal Dosimetry.

Table 5-6. Action levels for airborne radioactivity.

Years	Action level				Monitoring requirements	Remarks
	Alpha μCi/ml	Beta μCi/ml	Tritium μCi/ml	Gamma μCi/ml		
1945–1958	-	-			No information found	
1959–					>10% MPC	
1961					Bioassays when there is suspected uptake of material (Hasenkamp 1961)	Follow US AEC Rules
1971			2000– 100,000 dpm	-	Establish a Radiation Work Permit Area (O’Neal Burnett 1971)	
1994–1996	2.00E-12 (SNL 1994b)	2.00E-10 (SNL 1994b)			>2% of the DAC-hours (40 DAC-hours) or >100 mrem	10 CFR Part 835
1997– present	1.00E-12 (SNL 1999b)	1.00E-10 (SNL 1999b)				

Before 1992, bioassay was performed in response to suspected intakes (Hasenkamp 1961). Limits were based on ICRP Publication 2 (1959) maximum permissible concentrations (MPCs) and maximum permissible body burdens (MPBBs) (Hasenkamp 1961). Later limits were based on ICRP Publication 30 (ICRP 1979), including derived air concentration hour (DAC-hr) calculations; the requirements of 10 C.F.R. pt. 835 were being implemented (Potter 1994b, 1996).

In 1999, a memo-to-file describes the evaluation of the requirements of the routine bioassay program for TA-V. The tritium currently being produced by the pool was known to be 200,000 times less the concentration that would be required to produce a 1 DAC air concentration in the ACRR Hibay, therefore, the routine bioassay sampling for tritium was discontinued [24]. Routine semi-annual bioassay for uranium and annual whole body counting was continued (Culp 1999).

## 5.2 IN VITRO BIOASSAY

The *in vitro* bioassay program historically included the nuclides listed in Table 5-1. This section contains a detailed discussion of the selected radionuclides. Some of the bioassay was sent to outside laboratories. Formal procedures are available for radiochemistry results generated on the site after 1992. In most cases, MDAs are not provided for any of the results in the early years. MDAs can occasionally be inferred from the results, but these are inferences only [25]. A statement of work for the contract laboratories and a review of the program in 1994 list the desired radionuclides and the requested MDAs (Vosburg 1993a).

Current MDAs are available. These MDAs can be used for the program from 1992 to the present, but are not necessarily applicable to the 1945 to 1991 period. Because the Z Division was attached to LANL prior to November 1, 1949, the LANL MDAs do apply to bioassay results prior to November 1, 1949. Because of the MOU with LANL in the early years, LANL MDAs may be applicable for missed dose calculations. Also, since techniques used in the early years of operation were similar to those used at LANL and in the nuclear industry in general for uranium, tritium, and plutonium, MDAs from LANL techniques can be assumed [26].

### 5.2.1 Plutonium

Because of the nature of the operations at SNL, plutonium was not a major component of the potential intakes. Potential chronic missed intakes of plutonium should not be assumed unless there is evidence, through bioassay results or work history, that the individual was actually exposed to plutonium in a work place or field situations [27].

Plutonium bioassay was requested from the contract laboratory using alpha spectroscopy for the determination of  $^{238}\text{Pu}$  and  $^{239+240}\text{Pu}$  (alpha spectroscopy can not differentiate  $^{239}\text{Pu}$  and  $^{240}\text{Pu}$ ). Sample results from CEP between 1992 and 1994 are considered invalid and these results should not be used for determination of intake (Potter 1994a). However, if results of samples analyzed by CEP are found in a worker's record, the worker should be considered as monitored for the purposes of assigning missed or coworker dose [16].

Work with plutonium was done in gloveboxes or hoods [28].

#### **Respiratory Tract Absorption Type**

The absorption type assigned for plutonium isotopes is dependent on the chemical process used to produce the plutonium mixture. Table 5-7 lists absorption types (ICRP 1994b) in relation to compounds (Hempelmann, Richmond, and Voelz 1973). However, no information is available about the potential respiratory tract absorption type or aged mixture that was encountered at SNL. Excretion rates could differ from typical Type S compounds (Hammond, Lagerquist, and Mann 1968). Plutonium-238 could have been encountered as a high-fired oxide in the heat source technologies (Holley 1967).

Table 5-7. Plutonium respiratory tract absorption type.

Compound	Type	Comments
Nitrate, oxalate, sodium plutonyl acetate	M	
Metallic, fluoride, chloride, oxide	S	

Because the lung class W (Type M) of  $^{239}\text{Pu}$  and  $^{241}\text{Am}$  was considered limiting exposures to unknown mixtures, Class W (Type M) has historically been considered the default absorption at SNL for the purposes of dose assessment (Potter 1998a). However, a respiratory tract absorption type that is appropriate and favorable to the claimant should be selected.

In practice, much of the plutonium would be considered Type S because of the nature of the material. Material actually dispersed as a consequence of an explosion would have been subjected to high heat and therefore be in the oxide form. The particle size of the dispersed material was found to be a typically larger size than respirable particles (McConn 2006).

No other information about absorption types assigned to specific operations has been found. Therefore, if information is available about the compound involved, then the dose reconstructor should apply the absorption type as listed in Table 5-7 or select the absorption type most favorable to the claimant.

#### **Sample Collection Procedures**

Routine bioassay sample collection procedures were not well established at SNL before 1992. Most samples were collected on a protocol similar to the early LANL protocol which used three 1-L bottles for a 24-hr or simulated 24-hr sample. Spot samples might also have been taken [29]. Routine or special bioassay that was performed before 1992 was handled by Industrial Hygiene rather than Health Physics. Much of the bioassay monitoring was for nonradiological compounds associated with the processes at SNL; radiological bioassay was performed as part of the set of analysis performed

on the routine bioassay samples submitted by workers [30]. Archives of radiological bioassay have been found dating back to the early years of operations (Argall 2007a, b). The *Technical Basis Document for Internal Dosimetry* (Potter 1998a) states that urine bioassay was to be used to monitor routine intakes, but details about the frequencies and protocols associated with sample collection are not provided. Because exposure to plutonium at SNL was limited, extensive routine plutonium bioassay is not expected for most workers.

### Missed Intakes

Intakes of plutonium could occur from both acute and short-term chronic exposures. Chronic exposures might not be identified as incidents but can still result in a measurable burden of plutonium. However, long-term chronic exposures are not likely because of the nature of the tasks at SNL. If chronic intakes, favorable to the claimant are assigned in the absence of positive bioassay results, these intakes should only be assigned for workers in TA-V or for those workers participating in waste handling activities specifically involving plutonium [31].

The long-term excretion pattern of plutonium isotopes can permit plutonium intakes that produced bioassay results below the detection threshold in the early years to become detectable as the sensitivity of the analysis technique improved. The date of the intake might not be directly related to the last bioassay result below the detection level.

### Routine Sample Frequencies

In general, the need for air sampling, BZA samples, or bioassay appears to have been based on the amount of material in the process at the time and the perceived possibility of an intake [32]. Potter (1998a) discusses the use a NRC recommendation to apply a factor of  $1 \times 10^{-6}$  for the resuspension for loose radioactive material as the guidance for determining if the workday limits of 40 S-Derived Air Concentrations (DAC) or 0.02 S-Annual Limit on Intakes (ALI) are likely to be exceeded. While these exact criteria were not in place early in the history of SNL, the philosophy of monitoring only when there was a potential for exceeding the current guidelines appears to have been consistent throughout the history of the program [33].

According to a memorandum in 1993, individuals working with 2  $\mu\text{Ci}$  or more of plutonium would be required to participate in a bioassay program as directed by Internal Dosimetry (SNL 1992c).

A special sampling program would be initiated when a radiological incident occurs or a positive routine bioassay sample result is obtained that indicates the possibility of an unexpected dose of 100 mrem CEDE or more. Incident, Confirmatory, or Follow-up sampling protocols are initiated (SNL 1992c).

### Sample Analysis Procedures

Nondestructive analyses are currently performed on the SNL site. However, analyses requiring radiochemistry (e.g., plutonium) have been sent off the site since 1992 (Potter 1994a and Vosburg 1993b). However, according to conversations with a retiree and records in the SNL archives, beginning in 1949 samples were analyzed at SNL for total plutonium alpha (Argall 2007). These samples were prepared as deposited or electroplated samples in the Industrial Hygiene chemistry laboratories and transferred to the counting laboratory for proportional counting and, in later years, NTA or alpha spectrometry counting. Also, arrangements were made with Los Alamos and commercial laboratories to process samples (Hallman 1992). The responsibility for bioassay was transferred to the Radiation Protection Department in 1992. From 1992 any plutonium bioassay would have been performed off site. All sample results analyzed by CEP from 1992 to 1994 are considered invalid as previously discussed.

If plutonium bioassay results are found in dosimetry records, these results might be labeled either as total plutonium or as  $^{239}\text{Pu}$  (which contains  $^{240}\text{Pu}$ ) or  $^{238}\text{Pu}$ . Results labeled as  $^{239}\text{Pu}$  before 1967, the year when alpha spectroscopy was readily available in the industry, should be treated as total plutonium alpha unless also accompanied by  $^{238}\text{Pu}$  results [34].

Most *in vitro* bioassay samples are urine. Fecal and tissue samples are performed only if requested in special circumstances.

### Detection Sensitivities and Reporting Limits

MDAs are not available for historical samples, although MDAs in the “picocurie range” were achieved beginning in 1960 when the electroplated disks were counted using NTA (Argall 2007a). From 1992 to the present, the MDA has been 0.05 pCi/L for isotopic plutonium sent to an offsite laboratory (Potter 2006a). A 1993 statement of work for analytical services listed the required MDA for isotopic plutonium as 0.0017 Bq/L (0.05 pCi/L) (Vosburg 1993b). Results are currently reported as the calculated value, positive or negative. When not specified, the MDA is considered to be two times the detection level (or uncertainty of the blank). MDAs might be reported with results of samples before 1992. After 1992, a decision level ( $D_C$ ) equal to one-half the MDA might be reported with the sample results.

Because there was an MOU with LANL in the early years, after Z-Division was no longer directly controlled by LANL, the MDAs for plutonium analysis are included here and could be applicable. Table 5-8 contains the MDAs and reporting limits from LANL (Hallman 1992). Bioassay performed for employees of the Z-Division, during the time period that the Z-Division was part of LANL would have been performed using the applicable LANL procedures (Argall 2007a). Minimum detectable activity levels for bioassay performed at SNL before 1992 are not generally available unless listed with specific sample results. However, analytical techniques and instrumentation similar to those used at LANL were used at SNL; therefore, the MDAs would likely have been similar (Argall 2007a).

Plutonium results from LANL will most likely be reported in activity/24-hour sample. However, results provided by commercial laboratories are expected to be reported as picocuries per liter. Therefore, if volume units are not provided with the sample results and the results are not indicated as being provided by LANL or SNL, the dose reconstructor should assume the per-liter concentration.

### Plutonium Isotopic Ratios (Mixtures)

There is no definitive historical information on the  $^{240}\text{Pu}$ : $^{239}\text{Pu}$  atom ratios of SNL sources except for a ‘pure’  $^{238}\text{Pu}$  [35] source term (Holley 1967); and there is no information on how the ratios vary with time and location.

Exposure to heat-technology *pure*  $^{238}\text{Pu}$  ( $^{238}\text{PuO}_2$ ) might be suspected if positive  $^{238}\text{Pu}$  results are encountered when the associated  $^{239}\text{Pu}$  results are below the level of detection. A plutonium mixture that may be used when pure  $^{238}\text{Pu}$  from heat-source technology is encountered is:  $^{238}\text{Pu}$  (81%),  $^{239}\text{Pu}$  (15%),  $^{240}\text{Pu}$  (2.9%),  $^{241}\text{Pu}$  (0.1%), and  $^{242}\text{Pu}$  (0.3%) (Holley 1967). The disassembly of a  $^{238}\text{Pu}$  heat source in a glovebox was discussed in a SER logbook in 1972 (SNL 1974b). Table 5-9 lists the activity and weight ratios for heat technology plutonium mixture [36]. It is likely that pure  $^{238}\text{Pu}$  would be ‘fresh’ during processing in the heat source technologies. However, intakes of aged pure  $^{238}\text{Pu}$  may be possible during decontamination or decommissioning activities in areas where pure  $^{238}\text{Pu}$  was processed. Table 5-9 also includes the activity ratios necessary to calculate the appropriate mixture for aged ‘pure’  $^{238}\text{Pu}$ . Potential ingrowth of  $^{241}\text{Am}$  may also be determined from the ratios in Table 5-9 (Holley 1967). Weapons-grade (6%) and fuel-grade (12%) aged mixtures are described in Tables 5-10 and 5-11.

Table 5-10 lists activity and weight ratios as referenced for DOE sites for 6% weapons-grade plutonium. These ratios are based on the presumed age of the material for DOE sites. Table 5-11 lists the relative activities of plutonium isotopes and <sup>241</sup>Am, which grows in from <sup>241</sup>Pu (Gallaher and Efurd 2002), in 12% <sup>240</sup>Pu mixtures (Carbaugh 2003). In these tables, *aging* refers to the time since the <sup>241</sup>Am was separated from the plutonium.

Table 5-8. Plutonium bioassay sensitivity as listed in LANL procedures and reports.

Nuclide	Year(s)	Sample type <sup>c</sup>	Technique ("era")	MDA level	Reporting limit
				Unit/24-h sample <sup>a</sup>	Unit/24-h sample <sup>a</sup>
Pu-Total alpha	1944	U	Cupferron <sup>h</sup>	0.7 pCi (Clark 2005) <sup>b</sup>	>0.8 <sup>e</sup> pCi
	1945–1949	U	Cupferron <sup>h</sup>	0.16 pCi to 0.05 pCi <sup>d</sup> (1 dpm (0.45 pCi) DL (Moss 1990))	>0.8 <sup>e</sup> pCi
	1949–1/1957	U	Bi-phosphate/alpha counting	0.20 pCi to 0.07 pCi <sup>d</sup>	2 dpm or 0.9 pCi 0.4 pCi <sup>e</sup>
	1/1957–1965	U	Aluminum nitrate/NTA	0.03 pCi <sup>f</sup>	0.2 dpm or 0.09 pCi
	1966	U	ZnS	0.03 pCi <sup>g</sup> (0.07 dpm)	
Pu-239	1967–1991	U	Radiometric alpha spectroscopy (RAS) or pulse height analysis (PHA) only starting in 1971	0.03 pCi <sup>i</sup> (1 MBq)	
	1977–1981	F	RAS (PHA)	1 nCi/sample (less if Am-241 ratio known)	
	1981–1983	F	Phoswich detector, 4-cm sample thickness	0.4 nCi /sample or 400 pCi/sample (17-keV X-rays)	
Pu-238	1967–1971	U	RAS (alpha PHA)	0.03 pCi <sup>i</sup>	0.2 dpm/24 hr investigate
	1971–1976	U	RAS (alpha PHA)	0.03 pCi <sup>i</sup>	
	1977–1991	U	RAS (PHA)	0.03 pCi <sup>i</sup>	
	1977–1981	F	RAS (PHA)	0.4 nCi/ sample	
	1981–1983	F	Phoswich detector, 4-cm sample thickness	0.2 nCi / sample, (17-keV X-rays)	

- a. Unless otherwise noted.
- b. Not adjusted for potential chemical recovery.
- c. U=urine, F=fecal.
- d. Background count rate 1 cpm (changed to 0.1 cpm at some unknown time before 1957), 1,000-min count time, 50% efficiency, average recovery 82.3% ±19.4% (1945 to 1949) and 67% ±21% (Nov. 1949 to Jan. 1957).
- e. Results above these values were considered "high" (i.e., positive) and subject to statistical investigation. Source: Lawrence (1978).
- f. Source: Campbell et al. (1972); McInroy et al. (1991).
- g. Source: Moss et al. (1969).
- h. "A successful method of analyzing urine was developed in Jan. 1945 but could not be used as a routine test until a contamination free laboratory (ML Building) was ready for use in Feb. 1945" (Hempelmann 1945).
- i. Source: LASL (1978).

Table 5-9. 'Pure' <sup>238</sup>Pu isotopic mixture (Holley 1967).

Specific activity (Ci/g of mixture)	Weight fraction (fresh)	Fresh	5 yr	10 yr	15 yr	20 yr	30 yr
		0	5	10	15	20	30
Pu-238	0.81	1.370E+07	1.317E+07	1.266E+07	1.217E+07	1.170E+07	1.081E+07
Pu-239	0.15	1.048E+04	1.048E+04	1.048E+04	1.048E+04	1.047E+04	1.047E+04
Pu-240	0.029	6.352E+03	6.349E+03	6.345E+03	6.342E+03	6.339E+03	6.332E+03

Pu-241	0.008	2.061E+05	1.620E+05	1.274E+05	1.001E+05	7.871E+04	4.864E+04
Pu-242	0.003	3.933E+00	3.933E+00	3.933E+00	3.932E+00	3.932E+00	3.932E+00
Am-241	NA	0	1.463E+03	2.601E+03	3.484E+03	4.167E+03	5.094E+03
Pu-239+Pu-240	0.839	1.59E+04	1.59E+04	1.59E+04	1.59E+04	1.59E+04	1.59E+04
<b>Activity ratios<sup>b,c</sup></b>							
Pu-238/Pu-239		1,491.1	1,433.2	1,378.6	1,325.0	1,273.5	1,177.1
Pu-238/Pu-240		2,108.2	2,027.4	1,950.7	1,875.5	1,803.5	1,668.2
Pu-238/Pu-241		16.8	20.6	25.2	30.8	37.6	56.2
Pu-238/Pu-242		1.18E+06	1.13E+06	1.09E+06	1.04E+06	1.00E+06	9.27E+05
Pu-238/Am-241		0	2,278.2	1,232.7	883.8	710.3	536.8
Pu-238/Pu-239+Pu-240		873.4	839.6	807.8	776.5	746.4	690.1

- a. Time since separation of Am-241 from the plutonium mix.
- b. Calculate dose for Pu-239 and Pu-240 separately using these ratios.
- c. Pure Pu-238 was not on site at SNL until after the introduction of RAS for analysis of bioassay samples. Therefore, the activity ratios from total alpha plutonium are not applicable.

Table 5-10. Activity composition of reference weapons-grade 6% plutonium mixture.

Mixture designation: Years of aging <sup>a</sup> :	Fresh	5-yr	10-yr	15-yr	20-yr	25-yr	30-yr
0	5	10	15	20	25	30	30
Specific activity in mixture (Ci/g)							
Pu-238	8.56E-03	8.23E-03	7.91E-03	7.60E-03	7.31E-03	7.03E-03	6.75E-03
Pu-239	5.77E-02	5.77E-02	5.77E-02	5.77E-02	5.77E-02	5.77E-02	5.77E-02
Pu-240	1.36E-02	1.36E-02	1.36E-02	1.36E-02	1.36E-02	1.36E-02	1.36E-02
Pu-241	8.24E-01	6.48E-01	5.09E-01	4.00E-01	3.15E-01	2.48E-01	1.95E-01
Pu-242	1.97E-06	1.97E-06	1.97E-06	1.97E-06	1.97E-06	1.97E-06	1.97E-06
Am-241	0	5.83E-03	1.04E-02	1.39E-02	1.66E-02	1.87E-02	2.03E-02
Pu-239+240	7.13E-02	7.13E-02	7.13E-02	7.13E-02	7.12E-02	7.12E-02	7.12E-02
Pu-alpha	7.99E-02	7.95E-02	7.92E-02	7.89E-02	7.85E-02	7.83E-02	7.80E-02
Total alpha	7.99E-02	8.53E-02	8.96E-02	9.28E-02	9.52E-02	9.70E-02	9.83E-02
Activity ratios							
Pu-239+240:Am-241	N/A	12.2	6.87	5.13	4.28	3.80	3.50
Pu-239+240:Pu-238	8.33	8.67	9.01	9.38	9.74	10.1	10.5
Pu-239:Pu-240	4.24	4.24	4.24	4.24	4.24	4.24	4.24
Pu-241:Pu-239+240	11.6	9.09	7.15	5.62	4.42	3.48	2.73
Pu alpha:Pu-239+240	1.12	1.12	1.11	1.11	1.10	1.10	1.10
Pu alpha:Pu-238	9.33	9.66	10.0	10.4	10.7	11.1	11.6
Pu alpha:Am-241	NA	13.6	7.62	5.68	4.73	4.19	3.84
Pu-241: Pu alpha	10.3	8.15	6.43	5.07	4.01	3.17	2.50

- a. Time since separation of Am-241 from the plutonium mix.

Table 5-11. Activity composition of reference fuel-grade (12%) plutonium mixture.

Mixture designation: Years of aging <sup>a</sup> :	Fresh	5-yr	10-yr	15-yr	20-yr	25-yr	30-yr
0	5	10	15	20	25	30	30
Specific activity in mixture (Ci/g)							
Pu-238	1.71E-02	1.64E-02	1.58E-02	1.52E-02	1.46E-02	1.40E-02	1.35E-02
Pu-239	5.26E-02	5.26E-02	5.26E-02	5.26E-02	5.26E-02	5.26E-02	5.25E-02
Pu-240	2.72E-02	2.72E-02	2.72E-02	2.72E-02	2.72E-02	2.71E-02	2.71E-02
Pu-241	3.09E+00	2.43E+00	1.91E+00	1.50E+00	1.18E+00	9.29E-01	7.30E-01
Pu-242	3.93E-06	3.93E-06	3.93E-06	3.93E-06	3.93E-06	3.93E-06	3.93E-06
Am-241	0	2.19E-02	3.89E-02	5.22E-02	6.24E-02	7.03E-02	7.63E-02
Pu-239+240	7.98E-02	7.98E-02	7.98E-02	7.97E-02	7.97E-02	7.97E-02	7.97E-02
Pu-alpha	9.69E-02	9.62E-02	9.56E-02	9.49E-02	9.43E-02	9.37E-02	9.32E-02
Total alpha	9.69E-02	1.18E-01	1.35E-01	1.47E-01	1.57E-01	1.64E-01	1.69E-01
Activity ratios							
Pu-239+240:Am-241	NA	3.64	2.05	1.53	1.28	1.13	1.04
Pu-239+240:Pu-238	4.67	4.86	5.05	5.24	5.46	5.69	5.90
Pu-239:Pu-240	1.93	1.93	1.93	1.93	1.93	1.93	1.93
Pu-241:Pu-239+240	3.87E+1	3.05E+1	2.40E+1	1.88E+1	1.48E+1	1.17E+1	9.16
Pu alpha:Pu-239+240	1.21	1.21	1.20	1.19	1.18	1.18	1.17
Pu alpha:Pu-238	5.67	5.87	6.05	6.24	6.46	6.69	6.90

Mixture designation:	Fresh	5-yr	10-yr	15-yr	20-yr	25-yr	30-yr
Years of aging <sup>a</sup> :	0	5	10	15	20	25	30
Pu alpha:Am-241	NA	4.39	2.46	1.82	1.51	1.33	1.22
Pu-241:Pu alpha	31.9	25.3	20.0	15.8	12.5	9.91	7.83

a. Time since separation of the Am-241 from the plutonium mix.

The nature of the process work at SNL, however, involved research-grade plutonium mixtures usually considered to be fresh. While the plutonium mixture encountered in these processes should be assumed to be fresh, aged plutonium could have been encountered during building decommissioning and environmental restoration and in other areas where plutonium or plutonium contamination could have been in place for many years. Any <sup>241</sup>Am observed in lung counts performed years after the intake can be assumed to be the result of the in-growth of <sup>241</sup>Am from the <sup>241</sup>Pu in the mixture over time or the <sup>241</sup>Am in the initial plutonium mixture unless the incident report specifically indicates potential exposure to pure <sup>241</sup>Am or urine bioassay for <sup>241</sup>Am is found for the intake period. The most probable type of mixture encountered by the worker should be assumed to be a 6% plutonium mixture unless other information is available in the dosimetry records [37].

### 5.2.2 Americium

At SNL, <sup>241</sup>Am is usually encountered as a trace contaminant in plutonium. However, there is also a potential for exposure to pure <sup>241</sup>Am at SNL.

Americium-241 has always been included in the bioassay profile at SNL, but the specific source term driving the inclusion of <sup>241</sup>Am is not known [38]. However, documents have been found indicating that a pure <sup>241</sup>Am source term does exist and has existed at SNL, e.g., a spill of <sup>241</sup>Am was reported to have occurred on or about March 26, 1992 (SNL 1993). There is an indication that workers submitted samples for the americium bioassay program only if there was a potential for exposure to pure americium. Therefore, plutonium mixtures should not be inferred from americium bioassay results if no plutonium bioassay results are associated with the americium bioassay. Conversely, if the selected plutonium mixture, based on plutonium bioassay results, indicates the presence of americium, the absence of americium bioassay should not preclude the calculation of the dose from the americium contribution to the plutonium mixture.

Encounters with <sup>241</sup>Am could have occurred in relation to certain experiments or the handling of leaking calibration sources.

The potential for chronic exposure to pure <sup>241</sup>Am, or <sup>241</sup>Am in a plutonium mixture, at SNL is not routine. Potential missed chronic intakes of pure <sup>241</sup>Am should not be assigned unless there is evidence that the worker was at least intermittently exposed to <sup>241</sup>Am at SNL. This evidence would be from bioassay results and/or work histories [39].

### Sample Frequencies

Currently, no routine sampling programs are in place for <sup>241</sup>Am. A special sampling program would be initiated when a radiological incident occurs or a positive routine bioassay sample result is obtained that indicates the possibility of an unexpected dose of 100 mrem CEDE or more. Incident, Confirmatory or Followup sampling protocols are initiated (SNL 1992c).

### Minimum Detectable Activities

Since 1992, bioassay for <sup>241</sup>Am has been sent to offsite laboratories. All reported results have been listed as actual values, positive or negative. When not specified, two times the detection level is considered the MDA. The current MDA requested from offsite laboratories is 0.05 pCi/L (Vosburg 1993b; Potter 2006). No record of MDAs is available prior to 1992 but, because of the MOU with

LANL to perform bioassay analysis, LANL MDAs may be used if bioassay results without MDAs are found before 1992. Table 5-12 lists LANL MDAs.

Bioassay results listed as having been performed by CEP between 1992 and 1994 should be considered invalid (Ball et al. 1995). However, these results would indicate that the worker participated in a bioassay program for the purposes of assigning missed or coworker doses.

Table 5-12. <sup>241</sup>Am LANL bioassay techniques and sensitivities.

Sample type	Year	Method	MDA
Urine	1954–1957	Unknown	9.0E-01 pCi/24 hr <sup>a</sup>
Urine	1958–1982	Chemical extraction/ proportional counting	2.0E-01 pCi/24 hr <sup>b</sup>
Fecal	1977	Phoswich	4.0E-02 nCi/sample
Urine	1983 <sup>c</sup> –1991	Co-precipitation/alpha spectroscopy <sup>d</sup>	1.5E-02 pCi/24 hr
Fecal	1983 <sup>c</sup>	Am/Pu screening/Phoswich	1.0E-02 nCi/sample

- No MDA available, use derived investigation level; tolerance level 3.1 pCi/sample.
- Source: Milligan et al. (1958); method can carry over thorium, plutonium, curium, actinium, and neptunium. Exact end date to the start of this MDA is not known.
- Source: Gautier (1983); exact start date of the MDA is not known.
- Source: Inkret et al. (1998).

### 5.2.3 Tritium

A 1989 review of health physics and industrial hygiene functional areas listed limited tritium handling as one of the main potential source terms at SNL (Hyde 1989). Applying potential missed dose based on chronic intakes of tritium to workers whose duties involved work in areas involved with radioactive materials is reasonable.

Tritium was encountered in several forms: Tritiated water (HTO), tritiated gas (HT), organically bound tritium (OBT), and metal tritide (MT). Each form has unique characteristics. If the form is not indicated or the work area is not specific, then, with the exception of MTs, dose reconstructors should assume HTO, which is the form generally encountered (SNL 1990b).

Smear surveys of Building 642 in 1972 indicated significant tritium contamination on tools and target area parts. Some smears indicated more than  $5 \times 10^6$  dpm of tritium on parts and surfaces, especially those related to the target. Contamination was also found on the floors of the area. Buildings 805 and 806 are also a potential source of intakes [40].

A smear survey of the SER Cold Exhaust plenum indicated a maximum of 77 dpm/swipe (SNL 1989a). Tritium incidents have been recorded for accelerator areas [41].

The primary method of limiting uptakes of tritium was engineering controls of proper ventilation. Smearable tritium of 2000 dpm or less was considered a clean area. Smearable contamination of 2000 dpm to 100,000 dpm required that contamination control procedures be observed and RWP areas established. Above 100,000 dpm a Radiation Danger Zone was established (O'Neal Burnett 1971).

#### **Organically Bound Tritium**

The first approximation of the dose from OBT (labeled) compounds is the tritium in body water dose. However, the absorption, distribution, and excretion of tritium-labeled compounds are specific to the

chemical and physiological behavior of the particular compound. Specific guidance is available in NIOSH (2003) when the dose records indicate an exposure to tritium-labeled compounds.

Intakes of labeled compounds do not follow the same biokinetics as the ICRP Publication 68 defaults (ICRP 1994b). NIOSH (2003) discusses the Integrated Modules for Bioassay Analysis (IMBA) model for OBT.

### **Metal Tritides**

Tritium exposures in the form of MT aerosols were possible. The compounds include the chemical hydrides and dihydrides of hafnium, erbium, titanium, zirconium, and other metals.

In TA-I Building 891 the predominant form of tritium was erbium tritide. The ECF, Building 950, also presents the potential for exposure to MTs. MTs were used in neutron generators associated with the ACRR although the potential for intakes from these sources was demonstrated to be limited (SNL 1990b).

The telephone interview might provide indications that an energy employee had worked in locations where a potential for exposure to MT existed.

### **Other Sources of Potential Tritium Intakes**

In addition to the areas listed above and in Table 5-1, other noteworthy isolated incidents have occurred. Tritium swipe surveys have been performed consistently in many of the areas where there was a potential for tritium contamination. The MDA for a tritium swipe is approximately 55 dpm/filter (SNL 1995) and 13 dpm/filter (SNL 1999a). Table 5-13 lists these incidents including surveys that identified areas of contamination. (Not all incidents that have occurred at SNL are included in this table.)

### **Analytical Techniques**

All results since 1992 are likely to be reported as actual values, positive, negative, or zero (SNL 2005c, 2005d). The reported MDA for onsite tritium bioassay analysis is 1,000 pCi/L (0.001  $\mu$ Ci/L). A Statement of Work for analysis from an outside vendor requested a desired MDA of 9,000 pCi/L (Vosburg 1993b). The MDA before 1992 is currently unknown, but a report of all of the tritium results for 1989 indicates values of 0.01  $\mu$ Ci/L as the lowest value reported above 0.0. It appears that the analyses were performed at SNL (Hallman 1990). This agrees with the MDA reported by LANL at that time. LANL could have provided some analytical services to SNL before 1992. Typical MDAs reported by LANL before 1992 are listed in Table 5-14.

Bioassay results listed as having been performed by CEP between 1992 and 1994 should be considered invalid (Ball et al. 1995). However, these results would indicate that the worker participated in a bioassay program for the purposes of assigning missed or coworker doses.

### **Routine Sampling**

No information is currently available on historical routine sampling programs. At present, only waste handlers are on a routine tritium bioassay program. The SNL protocol assumed that persons working in areas where there was a potential for tritium exposure were monitored on an as-needed basis.

Table 5-13. Other sources of potential tritium intakes .

<b>Time</b>	<b>Description</b>	<b>Intake</b>	<b>Comments</b>
3/9/71	Tritium survey of 802/808 indicated measurable tritium on general surfaces.	Unknown	(SNL 1966)
6/22/73	Tritium contamination on work area surfaces Building 8002/ room 354/312, up to 7,142,000 dpm/smear on the source		(SNL 1966)

Time	Description	Intake	Comments
	and 574,000 dpm/smear on a tray.		
5/30/73	Memorandum discusses the potential for intakes while handling the 50 Ci of tritium in Bldg 806.		(SNL 1966)
8/7/73	Tritium contamination on work area surfaces Building 805/ room 124.		(SNL 1966)
1/15/76	Swipe surveys 6580/6585 levels to 26,000 dpm (neutron generator).		(SNL 1974b)
1989	Tritium bioassay results annual report indicate several workers with sample results >MDA.		(Hallman 1990)
1991	The front sights of the M-16 rifles carried by SNL security inspectors contained 3 to 6 mCi of tritium. It was discovered that several of these rifles had leaking tritium cartridges. As a result, several of the security inspectors had positive urine bioassays (Ball 1991).		Because of this incident security officers received tritium bioassay at both SNL and Tonopah
1991	Neutron generators containing <100 mCi of tritium in hydride form have been shown to be self-contained, and in 20 previous shots have produced no observable release of tritium.	None	ACRR committee (Townsend 1991)
1994	Thermocouples contaminated with tritium. 17,500 dpm removable.	None	Most tritium fixed on surface

Table 5-14. Tritium urine bioassay sensitivity levels (from LANL).

Time	Detection level	Reporting level	Counting method
1950–1951	5.18E4 Bq/24 h <sup>a</sup> (1 µCi/L)		Electroscope
1952–1953	5.18E4 Bq/24 h <sup>a</sup> (1 µCi/L)		Geiger-Mueller counter
1954–1957	5.18E4 Bq/24 h (1 µCi/L)		
1958–1968	5.18E4 Bq/24 h (1 µCi/L)		Internal Geiger-Mueller
1969 <sup>b</sup> –1987	1.04E3 Bq/24 h (0.02 µCi/L) <sup>b</sup>	5.18E4 Bq/24 h (1 µCi/L)	liquid scintillation counter (LSC)
1988–1992	5.18E2 Bq/24 h (0.01 µCi/L)	5.18E3 Bq/24 h (0.1 µCi/L)	LSC and 1 ml raw urine

a. Expected to be the same as 1954.

b. Source: Gautier (1983).

Historically, individuals likely to exceed the 0.25 rem (1/20 of 5 rem annual dose from AEC requirements) would require further investigation and control. This would correspond to a single acute dose producing a urine concentration at the time 0 of 23µCi/L. As an administrative control, individuals with urine bioassay of 20 µCi/L would be excluded from tritium work until they reached 10 µCi/L (O'Neal Burnett 1971).

A special sampling program would be initiated when a radiological incident occurs or a positive routine bioassay sample result is obtained that indicates the possibility of an unexpected dose of 100 mrem CEDE or more. Incident, Confirmatory, or Followup sampling protocols are initiated (SNL 1992c).

### Assignment of Tritium Dose from Monitoring at LLNL

Tritium monitoring results for SNL workers temporarily assigned to Lawrence Livermore National Laboratory (LLNL) were typically reported in dose rather than as bioassay results. The assigned annual dose can be entered into the POC calculation directly. However, this process overestimates the total measured tritium dose due to the fact that the historical assessment methodology would use a 12-day effective half-life for tritium and a beta linear energy transfer value of 1.7, both of which would lead to a higher dose than the current methodology. As a further overestimating assumption potential missed tritium dose of 0.01 rem/month can be assigned for each month during the employment period without positive tritium data when there was a potential for tritium intake. The

tritium bioassay data are available by request from LLNL when best estimate calculations are required [42].

#### 5.2.4 Uranium

Historically, uranium on the SNL site was DU, natural uranium (NU), or enriched uranium. Table 5-15 provides generic uranium conversion factors derived from Rich et al. (1988). Table 5-16 provides conversion factors for NU. A DU Briefing (Jow 1993) states that DU is considered  $\geq 99.75\%$   $^{238}\text{U}$  and  $\leq 0.25\%$   $^{235}\text{U}$  at SNL. Conversion factors differ among DOE sites because the fractional activity of the  $^{234}\text{U}$  differs in the enrichments. Thus, 0.25% enriched DU at SNL may have a different amount of  $^{234}\text{U}$  than the same enrichment at a different DOE site, depending on the source of the DU and the processing.

Table 5-15. Uranium conversion factors.

Enrichment percent	Fraction by activity			Fraction by mass			Total U pCi/ $\mu\text{g}$
	U-234	U-235	U-238	U-234	U-235	U-238	
0.1	0.225	0.010	0.765	0.00001597	0.001	0.99898	0.438
0.17	0.260	0.010	0.730	0.00002	0.0017	0.999	0.465
0.2	0.285	0.010	0.705	0.00002178	0.002	0.997978	0.476
0.3	0.340	0.010	0.650	0.00003	0.003	0.997	0.514
0.5	0.410	0.020	0.570	0.00004	0.005	0.995	0.591
0.7	0.480	0.020	0.500	0.00005	0.007	0.993	0.668
0.711	0.486	0.022	0.492	0.00005	0.007	0.993	0.672
0.72	0.486	0.022	0.492	0.00005	0.007	0.993	0.675
3	0.745	0.045	0.210	0.00019	0.030	0.970	1.57
90	0.970	0.029	0.001	0.010	0.900	0.090	62.1
93	0.972	0.027	0.001	0.010	0.930	0.060	65.1
95	0.973	0.026	0.001	0.010	0.950	0.040	67.2

Table 5-16. Natural uranium.

Isotope	U-234	U-235	U-236	U-238	Total
Weight fraction	0.0000537	0.0072	0	0.99274	
Specific activity pCi/ $\mu\text{g}$	0.33367	0.01557		0.33367	0.68291 <sup>a</sup>
Fraction of total activity	0.4886	0.0228	0	0.4886	

a. As listed in IMBA Version 1.0.42.

Jow also states that the particle size is typically nonrespirable (i.e.,  $> 10 \mu\text{m}$  AMAD). However, not all uranium encountered at SNL was of nonrespirable particle size [43]. DU was machined in the Toxic Shop in TA-I, Building 869 until c1985 and after that time, in Building 840. The machining process is capable of producing respirable aerosols (VanDenvender 1984).

According to Hyde (1989), DU machining operations represent one of the main potential source terms at SNL. Facilities that have been surveyed for DU contamination are Sled Track (Building 6742); Buildings 9920, 9939, and 9940; and SURTSEY (Building 6922) (Jow 1993). Most particles are considered to be  $\geq 1 \text{ mm}$ , which is much larger than the  $10\text{-}\mu\text{m}$  size considered to be respirable. However, oxides produced by high-energy events, such as penetration of hard armor, can produce some fraction of aerosolized ( $< 3 \mu\text{m}$ ) particles. Some explosion and implosion experiments (e.g., NEST) resulted in approximately 11% aerosolized particles. Reactor experiments resulted in less than 1% aerosolized particles. Areas of known contamination were and are posted and require monitoring, by frisking, before exit or direction by radiation protection before entry into the area.

Fuel-coolant interaction (FCI) experiments using corium thermite, which contains DU, ended at Buildings 9920 and 9940 in 1983. The corium thermite material used in these experiments was

prepared in Building 9941 using dust respirators as respiratory protection. While most of the material recovered following the experiments was >10- $\mu\text{m}$  in diameter, preparation of the material involved working with DU in the powdered form (Marshall 1993).

### Sampling Protocol

Historically and currently, there has been no routine sampling program for uranium. A special sampling program would be initiated when a radiological incident occurs or a positive routine bioassay sample result is obtained that indicates the possibility of an unexpected dose of 100 mrem CEDE or more [44]. Incident, Confirmatory or Followup sampling protocols are initiated (SNL 1992c). Bioassay sampling is initiated immediately following a suspected intake incident and continued for a "few" days [45].

Because there is no preservative used in the sample collection bottle, an acid wash of the bottle is required to collect uranium that could have plated out on the surface of the bottle during sample transport and storage [46].

Because no workers participate in a routine uranium bioassay program, there is no protocol for routine sampling on a Friday or Monday (Potter 2006c). Friday sampling would include the prompt-removal fraction and Monday sampling (after a weekend away from the site) would typically not.

According to a 1993 memorandum, individuals handling 10  $\mu\text{Ci}$  or more of DU would be required to participate in the bioassay program (SNL 1992c).

### Uranium Analysis Techniques

Techniques performed at SNL include fluorometry, kinetic phosphorescence analysis (KPA) and inductively coupled plasma mass spectroscopy (ICP-MS). Until 2003, uranium was analyzed by fluorometry using KPA or other fluorometric methods typical of the available instrumentation of the era. Fluorometry is based on the nonquantitative heavy-metal content in the urine. According to the Health Physics department appraisal (Hyde 1989), a quantitative analysis was not available to evaluate positive samples. In 2003, KPA was replaced by ICP-MS, which can provide isotopic information as well as total uranium.

### Analytical Sensitivity

The Industrial Hygiene Department was responsible for uranium bioassay from at least 1959 through 1991 (Potter and Sturgis 1993). A discussion with a retiree indicates that bioassay for uranium was being performed by the Industrial Hygiene Department as early as 1950. Results of these bioassays have been found in SNL archives. In 1991 responsibility was shifted to Internal Dosimetry [47]. Information on the analytical sensitivities for urine bioassay before 1992 is sporadic. Results identified for workers also include blank and standard results. Table 5-17 lists the known MDAs. Because there was an MOU for analysis with LANL in the early years, the LANL MDAs before 1975 can be assumed for the calculation of missed dose [48]. Samples may have been analyzed at contract laboratories other than LANL. After 1975, the MDAs in Table 5-15 are derived from analyses performed on site at SNL/NM or by a contract laboratory.

Bioassay results listed as having been performed by CEP between 1992 and 1994 should be considered invalid (Ball et al. 1995). However, these results would indicate that the worker participated in a bioassay program for the purposes of assigning missed or coworker doses.

Table 5-17. Routine uranium urinalysis detection levels.

Period	Method <sup>a</sup>	MDA	Decision level	Reporting level <sup>b</sup>
1949–1967	Fluorophotometric (DU or NU)	None listed <sup>c</sup>	50 $\mu\text{g/L}$	>100 $\mu\text{g/L}$

Period	Method <sup>a</sup>	MDA	Decision level	Reporting level <sup>b</sup>
1968–2/1976	Fluorophotometric (DU or NU)	4 µg/L <sup>c,d</sup> U		
3/1976–1978	Fluorophotometric (DU or NU)	1 µg/L <sup>d</sup> U		
1949–1954	Anion exchange/gross alpha counting (possibly used)	25 dpm/L		>100 dpm/L
1955–1971	Extraction/alpha proportional counting (U-234 alphas measured)	50 dpm/L <sup>c,e</sup>	50 dpm/L	>100 dpm/L
3/1971–1974	Extraction/alpha proportional counting	15 dpm/L <sup>c</sup>		>100 dpm/L
1975–2002	Total uranium – KPA or other fluorometric	0.1 µg/L		
2003–present	Total uranium – ICP-MS	0.1 µg/L (0.05 µg/L usually achieved)		
1993–present	Alpha spectroscopy (U-233, -234, -235, -238) outside vendor	0.0033 Bq/L (0.089 pCi/L)		

a. Method listed.

b. Exceeding reporting levels required investigation and evaluation (Lawrence 1984).

c. Lawrence (1984).

d. 50 µg/L considered positive indication of NU material in the body (Dummer 1958).

e. Specific for U-235 and U-233, 50 dpm/24-hr sample considered positive indication of enriched uranium in the body (Dummer 1958). Use 50 dpm/L as MDA because no other information is available.

### Selection of Absorption Type

If the compound to which the worker was exposed is unknown, the absorption type should be selected based on the value most favorable to the claimant. However, if the compound is known, Table 5-18 can be used to select the appropriate absorption type.

Table 5-18. Solubility class and absorption type assigned to uranium compounds (Rich et al. 1988).

Class D/Type F	Class W/Type M	Class Y/Type S
Uranium hexafluoride UF <sub>6</sub>	Uranium tetrafluoride UF <sub>4</sub>	Uranium aluminide UAl <sub>x</sub>
Uranyl fluoride UO <sub>2</sub> F <sub>2</sub>	Uranium oxide U <sub>3</sub> O <sub>8</sub>	Uranium carbide UC <sub>2</sub>
Uranyl nitrate UO <sub>2</sub> (NO <sub>3</sub> ) <sub>2</sub>	Uranium dioxide UO <sub>2</sub>	Uranium-zirconium alloy UZr
Uranyl acetate UO <sub>2</sub> (C <sub>2</sub> H <sub>3</sub> O <sub>2</sub> ) <sub>2</sub>	Uranium tetroxide UO <sub>4</sub>	High-fired uranium dioxide UO <sub>2</sub>
Uranyl chloride UO <sub>2</sub> Cl <sub>2</sub>	Ammonium diuranate (NH <sub>4</sub> ) <sub>2</sub> + U <sub>2</sub> O <sub>7</sub>	
Uranyl sulfate UO <sub>2</sub> SO <sub>4</sub>		
Uranium trioxide UO <sub>3</sub>		

### Environmental Uranium

New Mexico is known for high levels of NU in the soil and groundwater. However, some SNL workers could have lived in areas of particularly high NU concentrations ranging from 0 to 4 Bq/L (108 pCi/L) in 1992 and up to 6 Bq/L (162 pCi/L) in 2001 (Little, Miller, and Guilmette 2003a). These areas of high concentration are primarily in the Espanola area.

A 1992 study (Little, Miller, and Guilmette 2003b) listed the average drinking water concentrations for the area of Los Alamos, White Rock, and Santa Fe as 0.015 µg/L (0.01 pCi/L or 0.00037 Bq/L).

A study was conducted by SNL Internal Dosimetry staff to determine the background range and mean of the background range of urinary excretion for nonradiation workers at SNL (SNL 1993). That range was determined to be between 0.07 and 0.26 µg/L with a mean of 0.16 µg/L [49].

The bioassay results reported for individual workers do not have dietary uranium subtracted.

However, decisions to classify a sample as positive for occupational exposure were made based on the sample exceeding the expected urinary excretion rate of levels in nonradiation workers at SNL

(SNL 1993). According to Potter and Sturgis (1993), urine samples with results above background are reanalyzed by alpha spectrometry. Samples that remain above background are evaluated to determine if the uranium is natural or depleted. Re-sampling is initiated if the results are above 0.3 µg/L. Potter and Sturgis (1993) states that uranium bioassay had been performed at SNL/NM since 1959 [50]. As of 1993, SNL/NM had not identified any intakes of depleted uranium, only natural uranium had been identified. Natural uranium is not currently considered an occupational intake. Natural uranium was on the SNL/NM site from at least 1967 through the mid 1990's.

### 5.2.5 Fission and Activation Product Analysis

SPR, KIVA, SER, and other reactors in TA-V have operated throughout the history of SNL. Examples of fission and activation products that are contained in plant air samples, effluents, and primary coolant have been taken from various reports (SNL 1967, 1988, 1989a, 1989b). Activation products were also produced from photoactivation of metals in locations such as the PBFA II. Photoactivation of stainless steel parts produces long and short-half life radionuclides. The specific activities of these can be up to microcurie/gram quantities in localized areas. Handling can be delayed to permit the decay of short-half life products, such as isotopes of aluminum and copper. However, the ferrous isotopes (<sup>59</sup>Fe, half life 45 days) and other activation products, such as <sup>54</sup>Mn (half life greater than 300 days), can become an aerosol when these parts are machined. These radionuclides are listed in Table 5-19.

Table 5-19. Fission and activation product nuclides.

Type	Radionuclide	Limiting DAC µCi/ml
Fission gas	Kr-85m	
	Xe-133	
	Xe-135	2E-5
	Ar-41 <sup>a</sup>	
Fission products	Zr-95	
	I-131	2E-8
	Ru-103	
	Ru-106	
	Ba/La-140	5E-7
	Te-132	9E-8
	Y-94	
	Mo-99/Tc-99m	6E-5
	Zr-97/Nb-97	5E-7/ 3E-5
	Rb-89	
	I-132	3E-6
	I-133	1E-7
	I-134	
	I-135	
	Sr-91	1E-6
	Y-91m	7E-5
	Y-92	3E-6
Cs-136		
Cs-137		
Cs-138		
Ce-141	3E-7	
Ce-144		
Sb-124		
Activation products	W-187	
	Ag-110m	
	Ta-182	
	Mn-54	

Type	Radionuclide	Limiting DAC $\mu\text{Ci/ml}$
	Mn-56 <sup>b</sup>	
	Na-24 <sup>b</sup>	
	Cu-64 <sup>b</sup>	
	Cd-115 <sup>b</sup>	
	Al-28 <sup>b</sup>	
	Co-56	
	Co-57	
	Co-58	
	Co-60 <sup>b</sup>	
	Zn-65	
	Sb-122	
	Ni-57	
	Sn-117m	
	Sn-119m	
	Ta-182	
	Ta-183	
	As-76	
	Fe-59 <sup>b</sup>	
	Be-7	
Other	U-235	

a. Main contributor to activity at SER.

b. From SER (SNL 1967), PBFA II (VanDevender 1984).

Urine bioassay is currently performed on the site by gamma spectroscopy for fission and activation products (SNL 2005e, 2005f). A 1993 Statement of Work for an outside laboratory lists the required MDA for mixed fission products based on <sup>137</sup>Cs as 0.83 Bq/L (22.4 pCi/L) (Vosburg 1993b). The current urine bioassay MDA for <sup>137</sup>Cs is 11 pCi/L. Whole-body counting was also performed for fission and activation products that emit gammas. The MDAs are listed in Section 5.3.

An incident at the SPR II reactor, circa 1968, necessitated the evacuation of the control room personnel when fission products released from the pulse migrated to the control room. Subsequently the design was modified to prevent this leakage after the pulse [51]. The radionuclides identified in the airborne contamination were <sup>41</sup>Ar, <sup>91</sup>Sr, <sup>132</sup>Te, <sup>130,132,133</sup>I, and <sup>138</sup>Cs. Short-lived fission products will decay to longer lived fission products (Christiansen Golden 1968).

Bioassay results listed as having been performed by CEP between 1992 and 1994 should be considered invalid (Ball et al. 1995). However, these results would indicate that the worker participated in a bioassay program for the purposes of assigning missed or coworker doses.

### Strontium

Records of either routine or special <sup>90</sup>Sr urinalyses are very sparse. It appears that the records of <sup>90</sup>Sr analysis actually indicate that samples were sent to an outside laboratory. Strontium-90 dose can currently be reconstructed only when <sup>90</sup>Sr results are actually listed for an individual. Since 1992, the requested MDA for <sup>90</sup>Sr as total strontium analysis is 5 pCi/L (0.17 Bq/L). For isotopic strontium, <sup>90</sup>Sr, and <sup>89</sup>Sr, the desired MDA is 9 pCi/L (0.33 Bq/L). If results from a contract laboratory are available prior to 1992, these MDA can be applied as the requested MDAs if no other MDAs are listed. LANL did not perform strontium analysis on site regularly; therefore, it is unlikely that samples would have been sent to LANL for analysis.

A source term for <sup>90</sup>Sr and <sup>89</sup>Sr other than mixed fission products from reactors has not been identified.

Bioassay results listed as having been performed by CEP between 1992 and 1994 should be considered invalid (Ball et al. 1995). However, these results would indicate that the worker participated in a bioassay program for the purposes of assigning missed or coworker doses.

### **5.2.6 Accelerator Areas**

The accelerator areas such as HERMES III present unique nuclides in addition to potential tritium exposures. Air activation products are produced. These products are  $^{15}\text{O}$  and  $^{13}\text{N}$ . These nuclides have extremely short half-lives of 122.2 s and 9.97 min, respectively. Health physics practices restricted entry into areas where airborne contamination exists until the area had been ventilated for at least 20 min after shutdown. Therefore, occupational intakes of these nuclides are not expected to be of concern. Activation products can be produced in metallic materials as described for the PBFA II (VanDevender 1984).

Tritium from targets was of greater concern; tritium is discussed in Section 5.2.3.

### **5.2.7 Indoor Radon**

Indoor radon is considered an incidental exposure at SNL. Air samples are counted immediately after sampling and then 4 d later. In the reviewed reports and logs, most of the air filters were at background by the fourth day, which indicates that the initial activity was primarily radon daughters. However, the levels of radon are not considered above ambient levels encountered in general work areas [52].

In 1991, the DOE Indoor Radon Study was published (DOE 1991). Several areas of SNL Albuquerque were included in the study. Results ranged from 0.3 pCi/liter to 7.6 pCi/liter. The highest value was in Room B29-A of Building 869. The next highest room, the control room of Building 6578, had a concentration of 3.1 pCi/liter. All other locations sampled were 2.2 pCi/liter or less.

### **5.2.8 Other Limited-Exposure Radionuclides**

SNL has always been a center for research. As such, small-scale use (in terms of either the number of persons involved or activity of the source) of various radionuclides not addressed in previous sections has occurred throughout the history of SNL. Little or no documentation has been found on bioassay for these nuclides.

These nuclides should be assessed only when there is an indication that the worker had a potential for exposure to that nuclide.

#### **Tantalum-182**

Tantalum-182 is mentioned in several reports. Tantalum-182 was typically used as a sealed source in payloads, but the reports do mention suspected incidents of contamination and surveys to confirm the contamination (Kaye 1976, 1977). Sealed sources might typically have contained less than 1 Ci of material. These sources were fabricated at SNL (Riggan 1978). Contamination may result when tramp TaC, present on the outer surfaces of the nosetip is irradiated and  $^{182}\text{Ta}$  then also resides on the surface of the nosetip in addition to inside the payload. The source term is associated with ANT/payloads between 1974 and 1980.

Tantalum-182 is a gamma emitter and can be detected in gamma spectroscopy of urine samples or in a whole-body count.

### Selenium-75

Selenium-75 is mentioned in several reports. Selenium-75 was typically used as a sealed source in ANT/payload operations between 1974 and 1980. Quantities of more than 2 Ci are contained in these sources. Intakes are not expected unless there was an encounter with leaking sources. Selenium-75 is a gamma emitter and can be detected in gamma spectroscopy of urine samples or in a whole-body count.

### Thorium

Bioassay samples are sent off the site for analysis of thorium isotopes. The current requested MDA for alpha spectroscopy analysis is 0.05 pCi/L. In 1993, a Statement of Work to an outside laboratory lists a desired MDA of 0.0033 Bq/L (0.09 pCi/L) (Vosburg 1993b). Results are expected to be reported in units of picocuries per liter. However, Table 5-20 is available if conversion from micrograms per liter is necessary.

Table 5-20. Specific activity of thorium isotopes.

Isotope	Specific activity (pCi/ $\mu$ g)
Th-228	8.1946E+08
Th-230	2.0184E+04
Th-232	1.0966E-01

The source term for thorium operations at SNL began in 1959. Thorium-232, as an aged oxide, was containerized and sealed in 1996-1997.

### Neptunium-237

Neptunium-237 was a requested nuclide in the Statement of Work for an outside laboratory in 1993. The desired MDA using alpha spectroscopy was listed as 0.0017 Bq/L (0.05 pCi/L). However,  $^{237}\text{Np}$  is not mentioned as a source term for any of the areas listed in the site profile.

### Polonium-210

Incidents of a leaking  $^{210}\text{Po}$  source have been reported. The site profile mentions that  $^{210}\text{Po}$  is a radionuclide associated with neutron generators in TA-2, B-935 between 1969 and 1993. A leaking  $^{210}\text{Po}$  source was handled by a worker in 1968. A bioassay sample was submitted (O'Neel 1968). Other potential source terms are not known. If  $^{210}\text{Po}$  bioassay results (prior to 1960) are found in a worker's records, then the detection levels from LANL might be applicable because of the MOU for bioassay.

Detection limits for routine urinalysis are listed in Table 5-21. A procedure is listed for  $^{210}\text{Po}$  in urine in Gautier (1983). The 1958 detection levels can be considered to continue through the end of  $^{210}\text{Po}$  bioassay.

Table 5-21. Routine  $^{210}\text{Po}$  urinalysis detection levels.

Period	MDA	Recheck	Tolerance
1943-1952			440 dpm/24 hr <sup>a</sup>
1953			50 dpm/24 hr <sup>a</sup>
1954	10 dpm/L	100 dpm/L	500 dpm/L
1955-1957	0.1 pCi/L		
1958		100 dpm/L	500 dpm/L

a. Source: LASL (1979).

### 5.3 ***IN VIVO* MINIMUM DETECTABLE ACTIVITIES, ANALYTICAL METHODS, AND REPORTING PROTOCOLS**

In the early years, SNL had an MOU with LANL for performing whole-body and lung counting for workers. However, it is not known how many workers actually participated in the program. It appears that any whole-body counts that occurred were performed because of suspected intakes and not as part of a routine monitoring program for workers with a potential for intakes of fission and activation products.

*In vivo* counting equipment and techniques were developed in the late 1950s and have been in routine use at LANL for measuring X- and gamma-ray-emitting radionuclides since at least 1970 and possibly as early as 1960. There is some indication that some of the counts recorded between the beginning of the program in 1955 and the 1960s were performed for development of the program rather than actual suspected intakes. Counts during this period should be evaluated as closely as possible for validity in the dose reconstruction.

*In vivo* monitoring for uranium was performed by Helgeson in 1989 and possibly in 1990 and 1991 (HSS 1989). The results of these counts were above the detection level for many of the persons from TA-V who were counted (SNL 1992c). A positive bias has been suggested in the Helgeson results (Brake 1989).

The whole-body counter program was established at SNL in 1993. The whole-body counting system is the commercially produced Canberra Accuscan II system with two hyperpure germanium (HPGe) detectors. This is a vertical shadow shield with two scanning germanium detectors. The system is capable of measuring the entire torso or specific areas.

Currently, only RadCon Technicians and waste handlers receive annual whole body counts to confirm the adequacy of the personal air sampling program (Potter 2006b).

#### 5.3.1 **Whole-Body Counters—LANL (1955 to 1992)**

The first whole-body counter to be used at LANL was the HUMCO I. This *human counter* became operational in 1955. The counter consisted of a large double cylinder with a liquid scintillation fluid (possibly trichloroethylene) filling the annular space between the cylinders. The scintillation fluid was viewed with an array of 5-in. photomultiplier tubes on the outside wall of the cylinder. The individual was placed inside the count chamber. The count rate was compared to the background count rate. The system typically used two energy windows, one for  $^{40}\text{K}$  (1 to 2 MeV) and one for  $^{137}\text{Cs} + ^{40}\text{K}$  Compton counts between 0.5 and 0.8 MeV. The result was obtained by subtracting the contribution of  $^{40}\text{K}$ . The system was not used for photons below 100 keV. This system was used to screen individuals who might have been exposed to fission products at the reactors or in flyovers during weapons testing. It was also used to detect the *bremssstrahlung* from  $^{90}\text{Sr}$  intakes. The energy resolution of these counters was poor. When an elevation of the background in a region of interest was observed, the individual was referred for screening with either the shadow shield or full-shield 4- by 8-in. NaI(Tl) crystals (Healy 1970). The sensitivities of the NaI(Tl) crystal were approximately the same as those for the HUMCO II, except the count time was significantly longer.

The HUMCO II became operational in 1958. The HUMCO II was housed in a count room (SB-16) made of 7 in. of pre-World War II steel. The resolution was improved, but it remained a screening counter.

In 1970, an *in vivo* counter capable of measuring four separate regions of the body began operation (Vasilik and Aikin 1983). Twin Phoswich (CsI and NaI) detectors were placed over the lungs. The

two layers of the detector were capable of simultaneously, yet separately, monitoring chest burdens for 10- to 250-keV photons (NaI), for plutonium and uranium isotopes and  $^{241}\text{Am}$ , and for 200- to 2,000-keV photons (CsI) for a qualitative assessment of a variety of fission and activation nuclides. A planar HPGe detector monitored the region between 10 and 250 keV with excellent energy resolution and could be positioned over the liver or thyroid as needed. Finally, an HPGe (formerly a GeLi) detector was positioned under the prone subject. This detector was primarily for whole-body assessment. This system could both identify radionuclides and quantify the burdens. The twin Phoswich detectors were replaced by twin three-detector arrays of HPGe detectors in 1999. The Phoswich and germanium systems were operated concurrently during the period from 1998 to 1999. Two of the six HPGe detectors are used when a thyroid count is required.

### Minimum Detectable Activity and Decision Levels

During operations, an observed signal must exceed the  $L_C$  [decision level, formerly referred to as the maximum significant measured activity (MSMA)] to result in the decision that there is some “detected” activity in the lung, body, or other organ (Vasilik et al. 1984). The minimum detectable activity (MDA), formerly the minimum detectable true activity (MDTA), is the smallest amount of activity required to be in the lungs or organ so that a measurement of an individual can be expected to imply, correctly, that presence of activity with a predetermined degree of confidence. MDA (or MDTA) and  $L_C$  (or MSMA) values are listed in Table 5-22 for various years of operation. The MDA and  $L_C$  values for lung counting are summarized in Table 5-28. The values of MDA and  $L_C$  are calculated according to the theoretical developments of Currie and of Altshuler and Pasternack (Vasilik et al. 1984). No information is currently available on MDAs for the thyroid detector or for  $^{131}\text{I}$  or  $^{125}\text{I}$  in the whole-body count.

Table 5-22. Routine LANL whole-body counting detection levels.<sup>a</sup>

Period	Nuclide	$L_C$ (nCi)	MDA <sup>b,c</sup> (nCi)
1955–1958 <sup>d</sup>	Cs-137		8
	Sr-90 <sup>e</sup>		30
1959–1970 <sup>d</sup>	Cs-137		4
	Sr-90 <sup>e</sup>		30
1971–1984	Be-7	0.9	1.8
	Cs-134	0.9	1.8
	Cs-137	0.9	2.1
	Co-57	2.1	4.8
	Co-60	0.78	1.8
1985 to 1997	Tl-202	0.5	0.9
	C-11 (based on 511 keV) <sup>f</sup>	0.3	0.5
	Eu-152	2.2	3.3
	Co-58	0.5	0.9
	Co-56	0.5	0.9
	Hg-197	3.1	4.6
	Hg-195	2.5	3.7
	Hg-195m	1.8	3.2
	Hg-197m	3.8	6.0
	Hg-203	0.8	1.2
	Hg-193m	0.7	1.5
	Cs-134	0.5	1.1
	Os-185	0.6	1.1
	V-45	0.5	0.8
	Be-7	3.4	8.7
	Sc-46	0.5	0.9
Mn-54	0.5	0.9	
Cs-137	0.6	1.1	

Period	Nuclide	$L_C$ (nCi)	MDA <sup>b,c</sup> (nCi)
	Co-60	0.5	0.8
	Br-77	1.7	3.4
	Sb-124	0.4	0.8
	Ce-141 <sup>g</sup>	2.2	4.4
	Ce-144 <sup>g</sup>	12.1	24.2
	Cr-51	6.4	12.8
	Co-57	1.4	2.8
	Cu-67	1.5	3.0
	Fe-59	1.2	2.4
	Se-75	1.1	2.2
	Se-73	0.4	0.8
	Na-22	0.4	0.8
	Zn-65	0.8	1.6

- A listing of an MDA for a radionuclide does not necessarily mean that the nuclide was frequently encountered. The MDAs listed in the individual's results for a given count should be used if available.
- Based on 95% confidence of detection.
- MDA =  $L_C \times 2$ , unless otherwise specified.
- The HUMCO I and II systems were designed for screening subjects. Subjects found to have contamination levels above background were referred to the 4- by 8-in. NaI(Tl) detector, which had the same sensitivities with an extended count time.
- By *bremsstrahlung*.
- C-11 is a positron emitter with no photons. However, the 511-keV peak should always be present due to positron annihilation. The 511-keV peak can have interference contributions from other sources, including pair production interactions from nuclides with photon energies greater than 1,022 keV.
- Lower sensitivities might be available using the lung counter for certain nuclides if lung counting is appropriate to the dose reconstruction.

Whole-body and lung counting are the primary methods for determining intakes of fission and activation products. When both whole-body (*in vivo*) and *in vitro* results are available, whole-body count results are generally considered more sensitive and dose reconstructors should use them to determine intake.

Results found in bioassay records are generally reported in nanocuries unless otherwise indicated. Results listed as "NULL" indicate no detectable activity (NDA) rather than MDA. Results less than the  $L_C$  could be reported as NDA.

An *in vivo* count spectrum is not analyzed for a fission or activation product radionuclide unless a peak associated with that nuclide is visible in the spectrum. When that peak is visible, the suspected nuclide is added to the library and the spectrum is reanalyzed. Visual or non-library-driven software recognition of a peak can be subjective and not directly correlated to MDA or critical level calculations, especially with the broad peaks associated with scintillation detectors. For whole-body counts, it is not reasonable to assume that a worker was exposed to or is being monitored for all radionuclides potentially reportable simply because an MDA was determined and listed on the report.

In general, no information is available in the reports on the assignment of respiratory absorption type for specific fission and activation product nuclides.

### 5.3.2 Whole-Body Counter–SNL

The whole-body counting program was established at SNL in 1993. The whole-body counter is a Canberra Accuscan II shadow-shield two-detector HPGe whole-body counter. The counting protocol was established for sensitivities of  $^{137}\text{Cs}$ ,  $^{60}\text{Co}$ , and  $^{40}\text{K}$ . Sensitivities reported in 2005 are listed in

Table 5-23 below. These did not vary significantly over the years from 2001 to 2005. Results reported in 1994 are approximately one-half the MDA reported beginning in 2001. The routine count time has remained the same since 1993. During that period, the counting system was relocated and the analytical software was upgraded, but no explanation for the change in MDA was provided [53].

Table 5-23. Sensitivities for whole-body counting, 2005 (Reese 2006).

Radionuclide	MDA (nCi)
K-40	140
Co-60	8.5
Cs-137	12

Table 5-24 lists MDAs reported for a typical 10-min whole-body count. These are averages of MDAs reported for three workers evaluated in 1994 and the results of a single typical individual in 2005. The whole-body counts for these workers indicated positive results for only  $^{40}\text{K}$  in typical amounts.

Table 5-24. MDAs for whole-body counting (SNL 1994a, 2005g).

Radionuclide	1993–2000 MDA (nCi)	2001–present MDA (nCi)	Radionuclide	1993–2000 MDA (nCi)	2001–present MDA (nCi)
Be-7	52	88.7	U-235	Not reported	196 <sup>a</sup>
Na-24	5	8.98	Gd-153	Not reported	86.5
Mn-54	5	9.5	Ir-192	Not reported	15.9
Co-56	6	10.2	Tl-201	Not reported	94.3
Co-57	14	26	Hg-203	Not reported	16.7
Ni-57	6	12.4	Tl-207	Not reported	4,490 <sup>a</sup>
Co-58	5	9.45	Tl-208	Not reported	34.8
Co-60	5	8.79	Pb-210	Not reported	12,000 <sup>a</sup>
Zn-65	11	25.9	Pb-211	Not reported	355 <sup>a</sup>
Zr-95	7	17	Bi-212	Not reported	146 <sup>a</sup>
Mo-99	11	71.5	Pb-212	Not reported	34.3
Ru-103	6	11.9	Bi-214	Not reported	25
Ru-106	43	98.2	Pb-214	Not reported	33.1
Ag-110m	5	9.94	Rn-219	Not reported	162 <sup>a</sup>
Cd-115	17	25.8	Ra-223	Not reported	118 <sup>a</sup>
In-115m	12	17.5	Ra-224	Not reported	0
Sb-122	7	15.5	Ra-226	Not reported	531 <sup>a</sup>
Sb-124	5	10.9	Th-227	Not reported	132 <sup>a</sup>
Sb-125	17	38.1	Ac-228	Not reported	37.8
I-131	6	14.8	Ra-228	Not reported	61.4
Te-132	8	Not reported	Th-228	Not reported	390 <sup>a</sup>
Ba-133	9	20.5	Th-229	Not reported	216 <sup>a</sup>
Cs-134	5	12	Pa-231	Not reported	779 <sup>a</sup>
Cs-137	6	12	Th-231	Not reported	7,760 <sup>a</sup>
Ba/La-140	21	Not reported	Th-232	Not reported	103
Ce-141	21	43.7	Pa-233	Not reported	31.3
Pr-143	11	Not reported	Np-237	Not reported	1960 <sup>a</sup>
Ce-144	98	201	U-238	Not reported	522 <sup>a</sup>
Nd-147	55	81.8	Pu-239	Not reported	35,6000 <sup>a</sup>
Eu-154	24	53.2	Cm-243	Not reported	92.8
Eu-155	52	118 <sup>a</sup>	Na-22	Not reported	8.83
Ta-182	16	34.1	Cr-51	Not reported	132
Ta-183	26	220 <sup>a</sup>	Mn-52	Not reported	7.92
Am-241	98	255 <sup>a</sup>	Sr-85	Not reported	14.9
Fe-59	Not reported	20	Y-88	Not reported	8.07
Ce-139	Not reported	23.4	Ag-108m	Not reported	11.3
Eu-152	Not reported	78.2			

- a. The Accuscan II is a whole-body counter and is not optimized for low energy photons or for the geometry typically associated with chest counts. However, these nuclides are reported as MDA (or possibly as positive results) in individual whole body count reports. The reported MDAs should not be considered the primary method of assessing missed or bounding dose for these radionuclides when other types of bioassay results are available.

The nuclides listed in the MDA section of the whole-body count report do not necessarily indicate that the worker was potentially exposed to the entire list of nuclides. Before determining if potential chronic exposure should be assumed for any nuclide, it is necessary to determine if the worker's tasks presented the potential for intake. It is not reasonable to expect that an individual worker would be exposed to all of the nuclides listed in Table 5-24. In addition, for overestimates, Table 5-25 contains the list of nuclides that produce the highest dose to organs of interest based on the reported MDAs for that period. The organ of interest varies because the reported nuclides and the MDAs have changed over the history of the whole body counting program. When it is necessary to project missed dose, the nuclide can be used for the listed organs.

Table 5-25. Nuclides that produce the highest dose to organ of interest.

Organ	1971–1984 Nuclide/type MDA (nCi)	1985–1992 Nuclide/type MDA (nCi)	1993–1999 Nuclide/type MDA (nCi)	2001– present Nuclide/type MDA (nCi)
Adrenals	Co-60 / M (1.8)	Ce-144 / M (24.2)	Eu-154 / M (24)	Eu-154 / M (53.2)
Urinary bladder	Co-60 / M (1.8)	Ce-144 / M (24.2)	Ru-106 / F (43)	Ru-106 / F (98.2)
Brain	Co-60 / M (1.8)	Ce-144 / M (24.2)	Ru-106 / F (43)	Ru-106 / F (98.2)
Gall bladder	Co-60 / M (1.8)	Ce-144 / M (24.2)	Eu-154 / M (24)	Eu-154 / M (53.2)
Kidneys	Co-60 / M (1.8)	Ce-144 / M (24.2)	Eu-154 / M (24)	Eu-154 / M (53.2)
Liver	Co-60 / M (1.8)	Ce-144 / M (24.2)	Eu-154 / M (24)	Eu-154 / M (53.2)
Muscle	Co-60 / M (1.8)	Ce-144 / M (24.2)	Eu-154 / M (24)	Eu-154 / M (53.2)
Ovaries	Co-60 / M (1.8)	Ce-144 / M (24.2)	Eu-154 / M (24)	Eu-154 / M (53.2)
Pancreas	Co-60 / M (1.8)	Ce-144 / M (24.2)	Eu-154 / M (24)	Eu-154 / M (53.2)
Testes	Co-60 / M (1.8)	Ce-144 / M (24.2)	Ru-106 / F (43)	Ru-106 / F (98.2)
Thyroid	Co-60 / M (1.8)	Ce-144 / M (24.2)	I-131 / F (6)	I-131 / F (14.8)
R.B.M.	Co-60 / M (1.8)	Ce-144 / M (24.2)	Eu-154 / M (24)	Eu-154 / M (53.2)
Bone surface	Co-60 / M (1.8)	Ce-144 / M (24.2)	Eu-154 / M (24)	Eu-154 / M (53.2)
Stomach	Co-60 / M (1.8)	Ce-144 / M (24.2)	Eu-154 / M (24)	Eu-154 / M (53.2)
S.I.	Co-60 / M (1.8)	Ce-144 / S (24.2)	La-140 / M (21)	Eu-154 / M (24)
U.L.I.	Co-60 / M (1.8)	Ce-144 / S (24.2)	La-140 / M (21)	Mo-99 / S (71.5)
L.L.I.	Co-60 / M (1.8)	Ce-144 / S (24.2)	Ce-144 / S (98)	Ce-144 / S (201)
Skin	Co-60 / M (1.8)	Ce-144 / M (24.2)	Ru-106 / F (43)	Ru-106 / F (98.2)
Spleen	Co-60 / M (1.8)	Ce-144 / M (24.2)	Eu-154 / M (24)	Eu-154 / M (53.2)
Uterus	Co-60 / M (1.8)	Ce-144 / M (24.2)	Ru-106 / F (43)	Ru-106 / F (98.2)
ET	Co-60 / M (1.8)	Ce-144 / S (24.2)	La-140 / M (21)	Ce-144 / S (201)
Lung	Co-60 / M (1.8)	Ce-144 / S (24.2)	Ce-144 / S (98)	Ce-144 / S (201)
Colon	Co-60 / M (1.8)	Ce-144 / S (24.2)	Ce-144 / S (98)	Ce-144 / S (201)
ET1	Co-60 / M (1.8)	Ce-144 / S (24.2)	La-140 / M (21)	Mo-99 / S (71.5)
ET2	Co-60 / M (1.8)	Ce-144 / S (24.2)	La-140 / M (21)	Ce-144 / S (201)
LN(ET)	Co-60 / M (1.8)	Ce-144 / S (24.2)	La-140 / M (21)	Ce-144 / S (201)
BBsec	Co-60 / M (1.8)	Ce-144 / S (24.2)	Ce-144 / S (98)	Ce-144 / S (201)
BBbas	Co-60 / M (1.8)	Ce-144 / S (24.2)	Ce-144 / S (98)	Ce-144 / S (201)
Gonads	Co-60 / M (1.8)	Ce-144 / M (24.2)	Eu-154 / M (24)	Eu-154 / M (53.2)
Breast	Co-60 / S (1.8)	Ce-144 / M (24.2)	Eu-154 / M (24)	Eu-154 / M (53.2)
Heart wall	Co-60 / S (1.8)	Ce-144 / M (24.2)	Eu-154 / M (24)	Eu-154 / M (53.2)
Thymus	Co-60 / S (1.8)	Ce-144 / M (24.2)	Eu-154 / M (24)	Eu-154 / M (53.2)
bb	Co-60 / S (1.8)	Ce-144 / S (24.2)	Ce-144 / S (98)	Ce-144 / S (201)
AI	Co-60 / S (1.8)	Ce-144 / S (24.2)	Ce-144 / S (98)	Ce-144 / S (201)
LN(TH)	Co-60 / S (1.8)	Ce-144 / S (24.2)	Ce-144 / S (98)	Ce-144 / S (201)
Esophagus	Co-60 / S (1.8)	Ce-144 / M (24.2)	Eu-154 / M (24)	Eu-154 / M (53.2)

Table 5-26 contains a list of nuclides (with short half-lives) that do not represent a significant potential for chronic intakes. These nuclides were included in the whole-body counting analysis library

because they can be encountered at the accelerators. However, accelerator workers do not routinely participate in the whole-body counting program. These nuclides are not currently in the IMBA program. If it is necessary to evaluate positive results for these nuclides, then it will be necessary to contact dose reconstruction supervision for applicable parameters.

Table 5-26. Accelerator nuclides in whole-body count library.

Nuclide	Half life
Be-7	53.3 d
Co-56	78.7 d
Ni-57	1.5 d
In-115m	4.49 h
Cd-115	2.23 d
Sb-122	2.7 d
Te-132	3.26 d
Nd-147	11.0 d
Ta-183	5.1 d

### 5.3.3 Cesium-137 Intakes from Fallout

Most workers in the early days of whole-body counting had detectable activities of  $^{137}\text{Cs}$ . Most of this was attributed to fallout. Some workers had even higher levels of  $^{137}\text{Cs}$  from consumption of wild game. An  $L_c$  used to establish the difference between occupational and nonoccupational sources of  $^{137}\text{Cs}$  and other fallout radionuclide intakes has not been discovered in the records. In lieu of other information, the guidance in Table 5-27 can be used, if applicable.

Table 5-27. Mean body burdens of  $^{137}\text{Cs}$  from fallout in the United States (nCi).<sup>a</sup>

Year	Body burden	Year	Body burden
1953	0.27	1966	9.7
1954	1.1	1967	5.6
1955	2.2	1968	3.5
1956	4.3	1969	2.7
1957	5.1	1970	2.7
1958	6.5	1971	2.7
1959	8.1	1972	2.7
1960	6.8	1973	2.7
1961	4.6	1974	1.6
1962	6.0	1975	1.1
1963	11	1976	1.6
1964	19	1977	1.1
1965	16		

a. Source: NCRP (1987).

The  $^{137}\text{Cs}$  intake should be considered occupational if the same whole-body count detected other fission or activation products. It should also be considered occupational if a fission or activation product or radiostrontium urinalysis showed detectable activity and the sample was obtained in a reasonable time before or after the whole-body count or within the period between the previous and next whole-body counts. The reasonable time is based on the biological retention pattern of the radionuclide in the body.

All other fission or activation products identified in the whole-body or lung count should be considered occupational, unless specifically stated in the information provided for the individual.

National Council on Radiation Protection and Measurements (NCRP) Report 94 provides mean body burdens of  $^{137}\text{Cs}$  for the United States for the years most likely to produce interference with occupational whole-body count results (NCRP 1987). Those values are listed in Table 5-27. If no other fission or activation products are linked to the intake and the  $^{137}\text{Cs}$  result is less than the values in Table 5-27, the  $^{137}\text{Cs}$  result can be assumed to be due to fallout.

#### 5.3.4 Lung Burdens

Currently and historically, there is no chest counting system at SNL. With the exception of lung counts for uranium performed by Helgeson in 1989 and 1990, workers requiring lung counts would be sent to LANL. A 1992 internal memo indicates that this was done for at least one  $^{241}\text{Am}$  incident (Hallman 1992). Crites (1967) indicates that a worker was being scheduled for a "whole-body count" following a suspected plutonium intake. Therefore, the LANL chest counting system is discussed here. The report from the Helgeson counts is also discussed below.

##### **Chest Counting at LANL**

Lung burdens of  $^{239}\text{Pu}$ ,  $^{238}\text{Pu}$ , and  $^{241}\text{Am}$  were monitored at LANL using the Phoswich lung detectors beginning in 1970. The 59.5-keV gamma line of  $^{241}\text{Am}$  is used to determine the  $^{241}\text{Am}$  burden (50- to 70-keV region). If the isotopic ratio for a given intake is known, the  $^{239}\text{Pu}$  and  $^{238}\text{Pu}$  can be determined from the  $^{241}\text{Am}$ . Otherwise, the plutonium is determined from the U-L X-ray region. When  $^{241}\text{Am}$ ,  $^{239}\text{Pu}$ , and  $^{238}\text{Pu}$  are present, corrections for the contribution of the Np-L X-rays from the decay of  $^{241}\text{Am}$  to the 14- to 25-keV  $^{239}\text{Pu}$  and  $^{238}\text{Pu}$  region must be considered. The Phoswich detectors were eventually replaced by arrays of HPGe detectors, which greatly improved the energy resolution. Improved energy resolution permits the system to distinguish between gamma and X-ray lines that are closer together. However, because the U-L X-ray energies for the decay of  $^{239}\text{Pu}$  and  $^{238}\text{Pu}$  are the same, there is no way to differentiate between these two isotopes in a typical measurement. (If a significant plutonium burden is present, gamma rays of distinct energies permit differentiation of the isotopes.) Isotopic information about the exposure is used to determine the appropriate calibration factor.

Efficiency, and therefore the sensitivity level, varies for every count due to the effects of chest wall thickness on the attenuation of the 17-keV X-rays and the 59.5-keV gamma ray. Therefore, the MDA listed with the count, when available, should be used. The MDA and  $L_C$  values in Table 5-28 are nominal and are based on the calibration chest wall thickness (2.5 cm). These can be used for correlation with projected bioassay results. Chest wall thickness for the individual can typically be found on the White Card associated with the *in vivo* counting record. Chest wall thickness is estimated by weight:height ratios for routine counting and by ultrasound for special or positive counts. For lung (chest) counts, increases in chest wall thickness can increase the MDA for the individual count. LASL (1977) suggests that for large individuals, MDAs should be increased 50%. The dose reconstructor should use best judgment in determining the applicability of the listed MDAs for bounding missed dose projections. The MDA and  $L_C$  values for lung counting are summarized in Table 5-28.

The minimum intakes detectable using *in vivo* counting for americium, plutonium, or uranium are often much larger than the minimum intakes detectable through *in vitro* methods. Results or detection limits for *in vivo* methods might also be used to bound intakes determined from *in vitro* results of detection levels and not as the primary source of determination of intake or missed dose for these nuclides. Plutonium burdens from the 26 individuals from the original Manhattan Project have been calculated

Table 5-28. MDA  $L_C$  values for LANL lung counting (nCi).

Period	Radionuclide	$L_C$	MDA <sup>a</sup>
Extended count time 1977 <sup>b</sup>	Am-241		0.3
	Pu-238		10
	Pu-239		21
1980 -1984 <sup>c</sup> (Ennis 2003) <sup>d</sup> (1970 – 1979) <sup>e</sup>	Am-241	0.155	0.31
	Pu-238	11	22
	Pu-239	24	48
1984 (Vasilik et al. 1984) <sup>f</sup>	Am-241	0.16	0.32
	Pu-238	14	28
	Pu-239	30	60
1998 <sup>g</sup> to present <sup>h</sup> (Ennis 2003)	Am-241	0.1	0.2
	Am-243	0.1	0.2
	Pu-238	10	20
	Pu-239	31	62
	Th-234	0.85	1.7
	U-235	0.1	0.2
	Np-237	0.2	0.4
1998 to present Fission/activation products (Ennis 2003)	511 keV	0.1	0.2
	Be-7	0.35	0.7
	Ce-141	0.1	0.2
	Ce-144	0.25	0.5
	Co-56	0.1	0.2
	Co-58	0.05	0.1
	Co-60	0.1	0.2
	Cr-51	0.35	0.7
	Cs-134	0.05	0.1
	Cs-137	0.1	0.2
	Cu-67	0.1	0.2
	Eu-152	0.1	0.2
	Hg-203	0.05	0.1
	Mn-54	0.1	0.2
	Na-22	0.1	0.2
	Nd-147	0.1	0.2
	Os-185	0.05	0.1
	Ra-226	0.9	1.8
Sb-124	0.05	0.1	
Sc-46	0.1	0.2	
Se-75	0.05	0.1	
Tl-202	0.05	0.1	

- Assume chest wall thickness of 2.3 cm.
- As listed in 1977 Quarterly Progress Report, based on a 60-min count time and a person of average build, for a UPPU Club member (LASL 1977). The 60-minute count time is not typical of routine chest counting and therefore these MDAs should not be applied to missed dose calculations unless the counting time is known to be 60 minutes.
- Might be applicable to the startup of the Phoswich system in 1970; no other information available. For a 2000 second count time.
- Assume chest wall thickness of 2.5 cm.
- There is a reasonable correlation between the expected MDA for a 15- to 20-min count time, typical of standard *in vivo* counting beginning in 1970, and the 60-min count MDAs listed in 1977.
- $MDA = L_C \times 2$ ; recounts were performed if the results  $> L_C$ .
- Lung counter has 10-300 keV and 80- to 3,000-keV ranges, so a lower sensitivity for certain fission and activation products can be obtained.
- The phoswich detectors were replaced by an array of six planar HPGe detectors c 1998 (Ennis 2003).

by LANL to be 6 to 80 nCi since before 1946. These individuals have been followed for more than 50 yr. None of these individuals has had activity above background using *in vivo* techniques (Voelz et al. 1979; Voelz, Grier, and Hempelmann 1985) with the exception of one individual with a positive lung count for  $^{241}\text{Am}$  37 years later. This individual was suspected of an additional intake of plutonium containing  $^{241}\text{Am}$  in 1957.

Based on the discussion above, in-growth of  $^{241}\text{Am}$  from early intakes (1944 to 1945) should be expected to be negligible. Intakes in later years might result in  $^{241}\text{Am}$  in-growth or might have contained  $^{241}\text{Am}$  in the original intake.

Results in bioassay records should be assumed to be reported in nanocuries unless otherwise stated. Results listed as "NULL" indicate NDA rather than MDA. Results less than the  $L_C$  are marked as NDA in the database. The actual decision level values and counting errors might be available on the report for the individual.

All individuals who receive lung counts are monitored for  $^{239}\text{Pu}$  and  $^{241}\text{Am}$ . In recent years, routine  $^{235}\text{U}$  and  $^{234}\text{Th}$  (as  $^{238}\text{U}$ ) have been added to the LANL routine *in vivo* analysis library.

### **Chest Counting by Helgeson**

Documents indicate that in 1989, 1990, and 1991 a mobile chest-counting system was brought to the SNL site to perform chest counts of workers. Ninety-eight (98) whole-body counts, 60 *in vivo* lung counts for DU and  $^{235}\text{U}$ , and 18 lung counts for  $^{239}\text{Pu}$  were performed in 1989. The results were two whole-body positive counts for  $^{137}\text{Cs}$ , 59 lung positive counts for DU, and 13 lung counts positive for  $^{235}\text{U}$ . No positive counts were recorded for  $^{239}\text{Pu}$ . There is no indication of the MDA for these counts or a description of the counting equipment used (HSS 1989). Brake (1989) suggests that the Helgeson counting system may have been exhibiting a positive bias during that period.

### **5.3.5 Wound Monitoring**

No procedures for gamma counting of wounds have been found although wound counting is mentioned in the program overview and the Sandia National Laboratories Radiation Protection Internal Dosimetry Technical Basis Document (Potter 1994a).

### **5.3.6 Uncertainties**

#### ***In Vivo* Counting**

The listed counting uncertainties are typically reported as two standard deviations for counts performed by SNL.

Uncertainties associated with LANL chest counting are reduced by use of different calibrations for different chest wall thicknesses and use of ultrasound to measure chest wall thickness. A 1-sigma uncertainty of about 20% for americium and uranium values in chest counting, not including correction for interferences from bone and liver, is assumed. Uncertainties would be much higher for an individual with activity in the bone and/or liver. The uncertainty in lung activity estimates affected by contributions from activity in the liver and skeleton would likely range from 100% or more for levels near or below the MDA to 50% or more for activity above the MDA. The uncertainty in the estimate of chest thickness using the height:weight correction was at least 50%. The mathematical correction was made for routine counts. Special counts and counts with positive results were typically corrected using ultrasound chest wall thickness measurements (LASL 1977).

Based on the above discussion, the assumption provided in the *Internal Dose Reconstruction Implementation Guidelines* (NIOSH 2002b) is adequate and should be used; namely, the standard deviation is 0.3 times the MDA or reporting level, with the exception of chest counts, for which 0.5 times the MDA should be used (LASL 1977). 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). Actual tests for *in vivo* counts of phantoms show even smaller uncertainty, but 0.1 is appropriate for broad applications (Ennis 2003). If actual standard deviations or other indications of uncertainty are reported with a bioassay measurement result, the reported value should be used.

### ***In Vitro* Measurements**

The measurement uncertainty is not typically listed with the results. When uncertainty is listed, the confidence level is not normally listed.

### **Chelation Therapy**

At present, no documents have been found to indicate that chelation therapy has been used for workers at SNL. However, in the event that chelation therapy might have been used, there may be an indication in the individual dosimetry records. The chelation process causes the rate of bioelimination of the plutonium or americium to increase.

## **5.4 AIR CONCENTRATION IN SELECTED BUILDINGS**

### **Average Airborne Concentration Levels**

Studies have shown that room air concentration is not necessarily an accurate quantitative predictor of intake because of the variations in respiratory conditions and particle dispersion within an area. Even BZA samples are not always an accurate predictor of the amount of intake (Whicker 2004). However, maximum and average airborne concentrations can be helpful in establishing boundary conditions for intakes or for determining which locations had potential for inhalation intakes.

Some records of airborne contamination levels in buildings with high exposure potential have been found. The SERF Health Physics Handbook (Devlin 1964) lists the MPCU<sub>a</sub> as  $1 \times 10^{-09}$   $\mu\text{Ci}/\text{cc}$  for unidentified radionuclides in air assuming there were no alpha emitters or no  $^{90}\text{Sr}$ ,  $^{129}\text{I}$ ,  $^{210}\text{Pb}$ ,  $^{227}\text{Ac}$ ,  $^{228}\text{Ra}$ ,  $^{230}\text{Pa}$ ,  $^{241}\text{Pu}$  or  $^{249}\text{Bk}$ . Revision 01 to an Occupational Air Monitoring Assessment Procedure (Roberts 1996) was completed in 1996. It is not apparent when the original procedure went into effect. This procedure describes the parameters used to identify areas that will require air sampling, CAM alarm set point determination, air flow pattern studies, placement of air monitoring equipments, air sampling line design, and air monitoring in environmental restoration activities. A procedure, Airborne Radioactivity Sampling and Monitoring, Revision 01, appears to be implemented in 1995 (Rima 1995). Again, when the original procedures were developed or the protocols implemented are unknown.

The typical protocol for counting air filters has been to count the air filters immediately and again 2 to 4 d later to allow for the decay of the radon progeny. When second count results are available, these results are typically background or significantly reduced from the results of the initial count. Results from various locations and dates are listed in Table 5-29. Airborne concentrations were not available for all years of operation, especially for fission products. The reported results often do not include the volume of air through the filter; therefore, it is not possible to determine the actual airborne concentration. When available, average concentrations are calculated from general air samples as simple averages or averages obtained directly from SNL reports. Blanks in the table indicate that no information is currently available for that period. Action levels are the arithmetic mean plus 2 standard deviations at 95% confidence level.

Table 5-29. Airborne contamination results.

Building/area	Date	Max. alpha dpm	Max. beta dpm	Remarks
643	02/03/70	4.6		2-d decay (SNL 1970)
643	07/07/72	bkg		3-d decay
Rm 112 GBL	12/10/79	---	245	Unknown decay
Rm 112 GBL	10/10/80	---	70	Unknown decay
6580/ Room 106	11/25/80	bkg	8 cpm	4-d decay (SNL 1980a)
ACRR High Bay	09/25/89	372	1,128	Unknown decay (SNL 1990b)

Engineering controls to prevent intakes are part of the radiation protection program. Records indicate that efforts were being made to install engineering controls that would reduce airborne radioactivity in the SPR Building as early as 1961 (SNL 1980b). Air activity levels were reduced from  $>1 \mu\text{Ci}/\text{m}^3$  gross alpha immediately after a burst in 1961 to  $<0.1 \mu\text{Ci}/\text{m}^3$  when gypsum modifications were installed in 1964. Air activities 30 minutes after the burst were reduced from approximately  $0.5 \mu\text{Ci}/\text{m}^3$  to  $0.0005 \mu\text{Ci}/\text{m}^3$  after the gypsum modification.

#### 5.4.1 Respiratory Protection Program

Respiratory protection devices were provided during the Manhattan Project by the U.S. Army Chemical Warfare Laboratories. The M-9 mask with either the M-11 canister for gases and particulates or the M-14 canister for particulates was used beginning in 1946. The efficiency of the canister was 1 in 100,000 particles of dioctyl phthalate (AIHA 1963). This mask continued to be used as a standard until commercially produced masks were available. By the time the handbook on radiation protection (Dummer 1958) was published, a variety of U.S. Bureau of Mines-approved commercial respirators and self-contained breathing apparatus was available. Although the SNL Z-Division originated at LANL, the respiratory protection program at SNL is not clearly documented to reflect the LANL program. As late as 1993 (Marshall 1993), the use of "dust respirators" is listed as the protection for mixing powders containing DU. While documents have suggested that respirators were used during certain environmental restoration activities, protection factors should not be applied in the dose reconstruction because their use by specific individuals cannot be guaranteed.

Before the Tiger Team assessment in 1991, the respiratory protection program was administered by the Industrial Hygiene group (Gonzalez 1985). Upgrades to the respiratory protection program were proposed in the 1995 action plan in response to the Tiger Team Assessment.

#### **Contamination Surveys**

Records of area contamination surveys have been found. A resuspension factor of  $1 \times 10^{-6}$  is usually considered when determining the potential air concentration from the loose surface contamination [54]. Records of contamination surveys that have been performed were found and an example of the results are included in Table 5-30 and Table 5-31. Maximum values for alpha and beta results are listed. Contamination survey results varied from MDA to the maximum values listed.

Contamination limit action levels were listed as  $>500$  cpm of alpha per  $60 \text{ cm}^2$  or  $1 \text{ mR}/\text{h}$  beta-gamma fixed or any removable in 1961 (Hasenkamp 1961). A counting efficiency of 50% is usually assumed for alpha counting. A technique for taking smears was discussed in the SERF Health Physics Handbook (Devlin 1964). Smears were taken of  $1 \text{ ft}^2$  and limits were  $100\text{-}200 \text{ cpm}/1 \text{ ft}^2$ . This was equivalent to  $3000$  to  $6000 \text{ dpm}/\text{ft}^2$  given collection and counting efficiencies. Loose contamination was cleaned up promptly because of the adverse impact it could have on experiments.

The Safe Operating Procedure (SOP) for Building 869 Toxic Machine Shop, January 23, 1984, lists the limits for uncontaminated materials by  $<10$  cpm removable alpha or  $<200$  cpm removable beta-

Table 5-30. Contamination surveys in selected areas.

Area	Date	Alpha dpm (max)	Beta dpm (max)	Location	Remarks
SER/6520	1976	75	7,500	Rm 212	Table top/Inside sink (SNL 1977b)
SER/6520	1976	782	11,500		Hood/Berger Hut (SNL 1974a)
SER/6520	1977	14	20,000	Rm 212	Disassembly area/work surfaces (SNL 1976)
SER/6520	1979	10.5	15	Rm 104-121	(SNL 1979)
SER/6520	1978	16	1,900	Rm 212	(SNL 1989a)
SER/6520	1980	5	293	Rm 212	Open shelf (SNL 1989a)
SER/6520	1981	200	2,800		Parts on shelf (SNL 1989a)
TA-V 203	1986	249	112	203	Surfaces/ hot side (SNL 1986)
TA-1	1993	8350	2990	Industrial Hygiene Lab	average

Table 5-31. Contamination surveys for tritium in selected areas.

Area	Date	Tritium dpm/filter	Location	Remarks
TA-V	1999	<13.4	892	Unidentified surfaces
	1996	<30.3	STAAB	Various surfaces
	1994	< 55.3	887	Cubicle 64 surfaces
	1973	574,131	802/ 312	Plastic tray
	1973	306,151	802/ 312	hood
	1971	6	860/130	routine
	1967	8,569	802/ 356	Work bench (water leak) (Sanders 1967)
	1967	31,524	802/ 356	glovebox(water leak) (Sanders 1967)

gamma. The Toxic Machine Shop handled DU, fission, and activation products. Materials containing plutonium were not permitted in the Toxic Machine Shop (VanDevender 1984).

A procedure for contamination surveys was issued in 1993 (Shanks 1993). Results flags on smear survey reports circa 1994 indicate that actions were taken relative to results greater than or less than the critical level (approximately  $\frac{1}{2}$  the MDA) and greater than or less than the 2-sigma error. Reports from 1994 list action levels for smearable alpha of 20, 200, and 1000 dpm. The differences may be related to the type of survey. Action levels in 1995 through the present are listed as 20 dpm alpha and 1000 dpm beta-gamma for smearable contamination (SNL 1994c).

In an internal memo (Harrah and Hays 1984), concerns were raised about the lack of portal monitors to prevent the inadvertent egress of contaminated personnel from the Low Level Radioactive Waste Clean-up Facility (Bldg. 906, TA-II). Hand held instruments were available but the potential of transits to occur which would carry contamination outside of the building existed without the portal monitors. There was also concern that because there was a lack of standard laboratory equipment, workers would be inclined to bring in and take back their own equipment, which might be contaminated.

### Continuous Air Monitors

Documents indicate that CAMs were in use in many of the areas where airborne concentrations were expected. This included both alpha and beta/gamma CAMs in the Glovebox Laboratory (see Figure 5-1) in the SER Facility (SNL 1967). The SER Facility was designed in 1957 (Lane 1957). Documents also indicate that standard MPCs were being observed for appropriate contaminants (SNL 1967). Log books indicate that CAMs were not operational in some areas during various periods. For areas involving gloveboxes, however, other methods of detecting possible breaches of integrity of gloveboxes were in place (Gonzalez 1985). Gloveboxes are operated with negative pressure and inert atmospheres. Oxygen sensors in these gloveboxes detect the presence of the smallest leak [55]. In addition, CAMs were placed near the point of release where the air concentration would be at a maximum as a backup detection device. These CAMs were in addition to the general area CAMs.

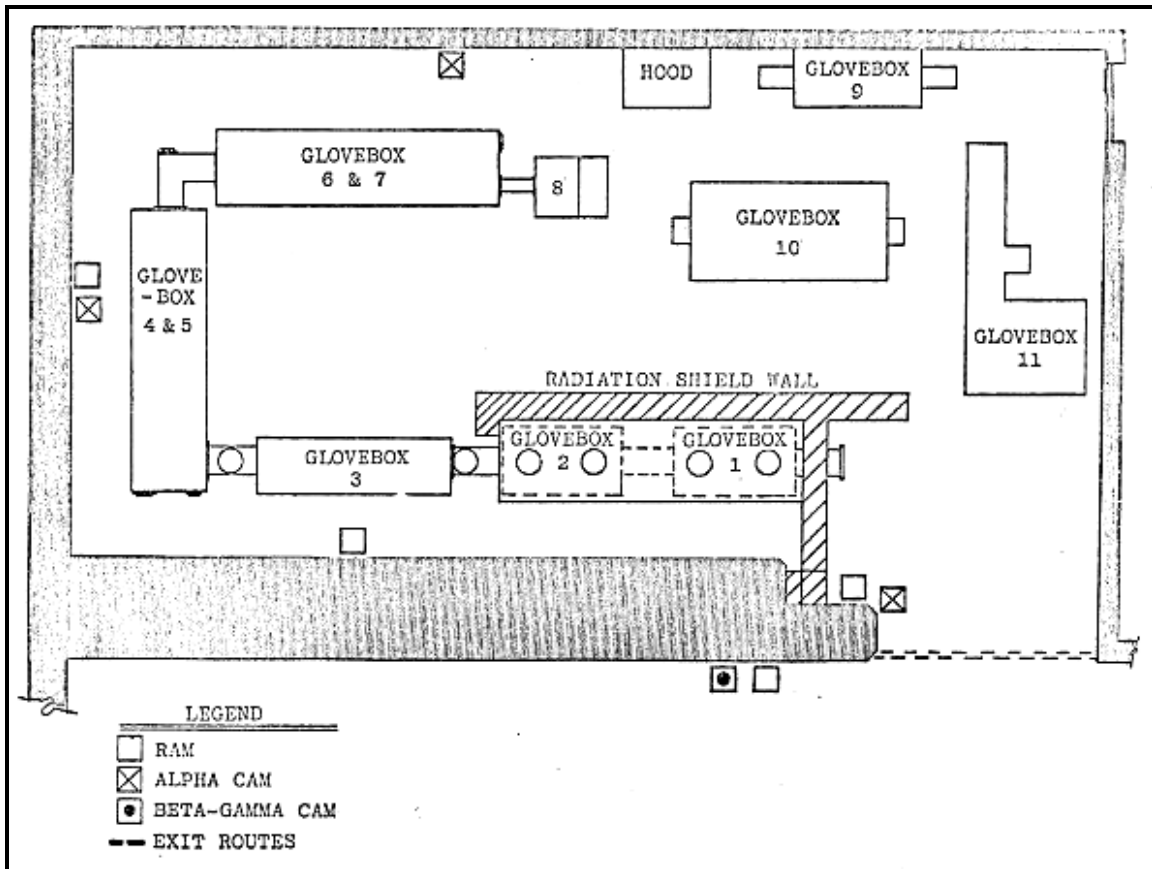


Figure 5-1. Layout of Glovebox Laboratory.

CAM alarms were historically used as an indication that room air concentration had changed and that respiratory protection was required. Whicker et al. (1997) found that the selection of the location of the CAM was critical to the reliability of the response. The typical location of CAMs at the ventilation exhaust point of a room was not considered optimal. Results also suggest that when a worker causes the release and is at or near the release point, the worker could be exposed for a significant period before a radioactive cloud reaches the CAM.

The above limitations suggest that, without other interventions, the possibility exists that workers could be exposed to intakes that did not trigger alarms.

#### 5.4.2 Maximizing and Best Estimate Intake Parameters

Intake parameters can be derived from airborne contamination levels for buildings with the highest exposure potential or highest intakes for various periods. Maximum and average airborne contamination levels for selected buildings are listed in Table 5-29. Examples of incidents and intakes are listed in Table 5-32. Average airborne contamination levels either are derived as simple averages or are reported as averages listed in reports. Simple averaging is assumed for SNL reports, but no information on the methods used to obtain these reported averages is available.

Particulate-filtering respirators were available and were often used from the beginning of the program. Therefore, ambient air concentrations might not reflect the actual breathing-air concentration of the workers.

Table 5-32. Reported exposure incidents and results.<sup>a</sup>

Date	Incident
06/04/67	Contamination of laboratory area Building 802, Room 360, water leaking on tritium-contaminated equipment, 2,440 dpm/ft <sup>2</sup> (Sanders 1967).
11/19/67	Positive bioassay for Pu-239 from individual in Organization 9311 (Bldg. 849, trailer B-60), 1.5 times maximum permissible body burden (Crites 1967).
3/1968 to 4/1968	Positive bioassay for tritium for workers, maximum 18 µCi/L. No incident discussed (Crites 1968).
03/13/68 to 03/25/68	Div 4233, individual with 27 µCi/L tritium urine sample last sample 11.7 µCi/L (Crites 1968).
03/08/68	Pu-238 "spill", particle found on CAM air filter in Tr B-60 Hot Lab (Hudson 1968).
11/22/68	Leaking Po-210 source transferred between laboratories. No intakes confirmed (O'Neal 1968).
01/14/70	Pu-238 spill, a leak discovered in a glovebox during cleaning. Contamination of the area in Bldg 643 (SNL 1970). Employees submitted bioassay samples.
9/21/83	Be-7, Zn-65 and Ag-110m found on Radeco air filters Rm 100 and 101 Bldg 6580 (SNL 1985).
04/12/91	Leaking neutron calibration device containing gaseous tritium in Area. Approximately 150 concerned employees requested bioassay.
03/26/92	Am-241, spill location unknown, urine bioassay 1.38E-1 pCi/sample, listed as false positive.
3/30/93	Am-241 contamination >10,000 dpm on tray in Bldg 869 Rm 16, area not posted as Controlled Area. Tray in use for 10 years (Byers 1993).
Unknown	Pu cladding breached on pit (McConn 2006).

a. While an attempt has been made to report only incidents with quantitative results, some incidents, for which only qualitative comments were available in the records, have been included in the list. Information is not available for every year of operation.

## 5.5 UNMONITORED WORKERS

Historically, routine bioassay monitoring programs were not well established although discussions with retirees indicate that workers with a potential for exceeding regulatory dose limits were monitored as required. In the pre-1970 years approximately 100 workers were monitored annually, mostly for tritium. Monitoring was performed in response to suspected intakes (Hasenkamp 1961). However, many of the areas at SNL were involved in work with nonradioactive materials where the potential of incidental intakes did not exist. When a worker does not have monitoring results in the DOE records, their work locations should be reviewed. These locations should be compared to the radioactive materials listed for locations in previous sections. Job titles are not necessarily the most accurate indicator of potential for exposure.

Also, the potential exists that not all of the early bioassay records have been transferred to the current database. These early records were transferred to microfilm or microfiche beginning in 1967 (Argall 2007b) and a summary of the resultant dose (or 0.00 rem if the results were <MDA) was entered into an electronic database. However, this information may not have been captured in the current database and thus not provided with the dose records supplied by DOE. If the job title or work locations indicate that the worker should have been monitored, bioassay results might be found in a search of the pre-1992 archived records.

Dose reconstructors should note any indication that a worker might have accompanied their equipment to a test site or a weapons storage site where the potential for intake could have been higher than at the SNL site where the design or fabrication took place. Also, SNL workers assigned to weapons storage sites may have handled unsealed materials (McConn 2006) and did not participate in bioassay programs at the storage site or immediately upon return to SNL (Argall 2007a). Dosimetry records from other sites are typically not included in the dosimetry records supplied for the SNL site. If records are not found, but exposure at other sites is indicated, then these records can be requested or site technical basis documents can be consulted.

### 5.5.1 Unmonitored Tritium Dose

Chronic intakes of tritium are possible and not uncommon as shown in the summary of 1989 tritium bioassay results by Hallman (1990). Tritium bioassay was performed from 1949 (Argall 2007b). While the majority of monitored workers had sample results less than 1  $\mu\text{Ci/L}$ , several workers had bioassay results between 10 and 74  $\mu\text{Ci/L}$  in 1989. These results and subsequent follow-up bioassay exhibited the expected biological half-lives; therefore, valid intakes are assumed. These maximum results can be used to determine overestimating doses for tritium for unmonitored workers. However, there is no evidence to suggest that workers with potential for tritium intake were not monitored.

### 5.5.2 Unmonitored Uranium Dose

The potential for intakes of uranium existed although SNL considered most uranium to be of the nonrespirable particle size. Some urine bioassay results have been found in the archives as early as 1975. Other results exist on microfilm and microfiche (Argall 2007a, b).

Argall (2007a, b) indicates that uranium bioassay may have been performed for some workers as early as 1949 or 1950.

In 1989, Helgeson performed lung and whole-body counts at SNL for uranium (depleted and  $^{235}\text{U}$ ),  $^{239}\text{Pu}$ , and  $^{137}\text{Cs}$ . Of the 60 counts, 59 were positive for DU and 13 were positive for  $^{235}\text{U}$ . A positive bias for Helgeson uranium lung counts has been suggested by Brake (1989). The maximum burden of DU listed in the summary report was 11.4 mg (HSS 1989). The individual results list a maximum of 8.3 nCi DU. The results for  $^{235}\text{U}$  are between 0.0 and 119 pCi. There is no indication of what the MDAs was for the lung counts. However, values were reported as 0.0 with the smallest result above 0.0 being 1.3 nCi. Maximum results can be used for overestimating assumptions.

Workers handling weapons assembly at weapon's storage sites may have been exposed to unsealed uranium, especially before 1957 (McConn 2006, Argall 2007a).

### 5.5.3 Unmonitored Fission Product Dose

It is not reasonable to assume that a worker was chronically exposed to all of the fission product and activation product nuclides. Therefore, when assigning unmonitored dose the dose reconstructor can refer to the fission product discussion in Section 5.2.5 and Table 5-25 to select the nuclide that produces the highest dose to the applicable organ.

Logs indicate that CAMs were in place in many areas as early as the 1950s (SNL 1967). Maximum allowable concentrations and later MPCs were being observed. An MPC for unidentified radionuclides in air of  $1 \times 10^{-9}$   $\mu\text{Ci/cc}$ , assuming no alpha emitters or beta emitters of  $^{90}\text{Sr}$ ,  $^{129}\text{I}$ ,  $^{210}\text{Pb}$ ,  $^{227}\text{Ac}$ ,  $^{228}\text{Ra}$ ,  $^{230}\text{Pa}$ ,  $^{241}\text{Pu}$ , and  $^{249}\text{Bk}$  present in the air, was in effect at the SERF in 1964 (Devlin 1964). Documents indicate that monitoring was based on the potential for exceeding an administrative limit. After 1992, persons were placed on bioassay programs if they were expected to have intakes that exceeded 40 DAC-hr (Potter 1998b). Administrative limits before 1992 were based on 10% of the MPC-hr and later, these were based on the potential for exceeding 100 mrem (SNL 1992c).

As bioassay sensitivities and respiratory protection equipment improved, the potential for intakes decreased [56].

## 6.0 OCCUPATIONAL EXTERNAL DOSE

### 6.1 DOSIMETERS USED

Beta/gamma and neutron dosimeters used at Sandia National Laboratories NM (SNL) over the years are described in Tables 6-1 and 6-2. SNL switched from film to thermoluminescent dosimetry after their supplier of dosimetry film discontinued production in November 1969 (Tucker 1970). Stockpiling of film enabled use of film dosimeters through approximately April 1971.

Table 6-1. Beta/photon dosimeters.

Period	Beta/photon dosimeters used
1949 to 1958	<u>Film Badge 1</u> : Metal holders were used that had a brass clip that covered one end of the film packets. The brass clip was intended to attenuate beta rays but not stop gamma rays (Kingsley 1953a). This holder was also issued as a wrist badge.
1959 to April 1971	<u>Film Badge 2</u> : Plastic holder had 4 windows: open window, 0.035-in. Al filter for beta/gamma, Tungsten/Cd and Sn filters for thermal neutrons. Used DuPont 558 film packets for the beta/gamma exposures based on calibrations with Co-60 and 70 keV X-rays. These packets contained DuPont 519 film (stated range 30 mR to 10 R) and 1290 film (stated range 10R to 3000 R) (Drake 1959, Tucker and Drake 1960).
May 1971 to ~1988	<u>Harshaw Model 2271</u> : The first TL badge had an open window and a 0.035-in. Al filter. The TLD card consisted of two TLD-100 elements, each also 0.035-in. thick. (Tucker 1970, Thompson et al. 1972, Kingsley 1971, Thompson et al. 1988).
~1989 to ~1994	<u>Harshaw Model 8801 (7776-1141) cards in Model 8812 holder</u> (Bradley 1993a, b, c; Friedman 1991; Rhea 1991; Ward 1994) <ol style="list-style-type: none"> <li>1: 0.015-in. thick TLD-700 under 600 mg/cm<sup>2</sup> ABS plastic (deep dose)</li> <li>2: 0.015-in. thick TLD-700 under 242 mg/cm<sup>2</sup> ABS, 0.004-in. Cu (low-energy X-rays)</li> <li>3: 0.0036-in. thick TLD-700 under open window, 0.0025-in. Mylar (shallow dose)</li> <li>4: 0.015-in. thick TLD-600 under 600 mg/cm<sup>2</sup> ABS plastic (neutron dose)</li> </ol>
~1995 to present	Harshaw/Bicron EXTRAD put into use 2nd quarter 1997 for extremity dose. <u>Harshaw Model 8802 (7776-1161) or 8801 cards (see above) in Model 8812 holder</u> <ol style="list-style-type: none"> <li>1: 0.015-in. thick TLD-700 under 600 mg/cm<sup>2</sup> ABS plastic (deep dose)</li> <li>2: 0.015-in. thick TLD-700 under 242 mg/cm<sup>2</sup> ABS, 0.004-in. Cu (low-energy X-rays)</li> <li>3: 0.006-in. thick TLD-700 under open window, 0.0025-in. Mylar (thicker shallow-dose chip than in the Model 8801)</li> <li>4: 0.015-in. thick TLD-600 under 600 mg/cm<sup>2</sup> ABS plastic (neutron dose)</li> </ol> (Walker 1996, Walker 1997b, Bradley 1993d, Bradley 1994, Bradley 1995a, Bradley 1995b)

Table 6-2. Neutron dosimeters.

Period	Neutron dosimeters used
1945 to 1958	No neutron dosimeters were used.
1959 to April 1971	<u>Film Badge 2</u> : In a plastic holder had four windows; ones with Cd and Sn filters were used to measure thermal neutrons. Used DuPont film for thermal neutron dose and Kodak Personnel Monitoring Film, Type A film (NTA) for fast neutron dose based on calibrations with a Van de Graaff accelerator at energies of 1, 5, and 14 MeV. (Drake 1959, Tucker and Drake 1960).
May 1971 to ~1973	The first TL neutron badge consisted of three LiF elements: two LiF-600 that were sensitive to thermal neutrons and one LiF-700 that was insensitive to neutrons but was used to subtract any gamma contribution. The LiF elements were placed behind tin and cadmium filters to support determination of incident and scattered thermal neutrons (Tucker 1970, Kingsley 1971).
~1973 to ~1990	First dedicated neutron dosimeter incorporated two cards, each with a TLD-600 and a TLD-700 element. One TLD-600 and TLD-700 were enclosed in a borated polyethylene pouch and one TLD-600 and TLD-700 were uncovered. The holder had an open window and three 0.035-in. Al filters. Deep and shallow measurements were made with the TLD-700 elements and bare and boron-filtered measurements of neutrons with the TLD-600 elements to discriminate energies near thermal (Thompson et al. 1988).
July 1984	An albedo TLD was added for neutron dosimetry (Thompson et al. 1988).



The following relationships were used starting around 1971 to calculate compliance dose quantities (Ward et al. 1994):

$$\begin{aligned} \text{Skin dose} &= \text{Shallow Dose} + \text{Deep Dose} + \text{Neutron Dose} \\ \text{Whole Body Dose} &= \text{Deep Dose} + \text{Neutron Dose} \\ \text{Extremities Dose} &= \text{Shallow Dose} + \text{Deep Dose} + \text{Neutron Dose} \end{aligned}$$

Electronic files of annual dose records received from SNL-NM for years since 1986 include deep dose, neutron dose, eye dose, shallow dose, extremity dose, CEDE, and TEDE (Widner 2006).

### 6.3 UNMONITORED PHOTON DOSE

Detailed dose data from SNL facilities have been obtained for 1949 to 2005 in forms that support calculation of lognormal probability statistical parameters desired for characterization of annual dose distributions for use in assignment of unmonitored photon doses (Widner 2006). Tables 6-4 and 6-5 summarize the respective lognormal probability statistical parameters for SNL-NM dosimeter results for penetrating and nonpenetrating dose that are equal to or exceed 50 mrem for years of record between 1949 and 2005 (Widner 2006). These data are useful to examine trends in the recorded doses. It should be noted that the reported doses that are the basis of these tables have not been corrected for potential missed doses, which is crucial to estimating potential dose for unmonitored workers.

Table 6-4. SNL-NM worker penetrating dose statistics, 1949 to 2005 (Widner 2006).

Year	SNL-NM recorded gamma dose data			Lognormal fit		
	No. of workers reported gamma dose $\geq 50$ mrem	Gamma dose (mrem)		Gamma dose (mrem)		GSD
		Mean	Maximum	Median	95%-ile	
1949	10	110	420	Use 159	-- <sup>a</sup>	-- <sup>a</sup>
1950	41	240	1,850	159	588	2.21
1951	84	200	3,000	134	452	2.09
1952	68	770	5,100	279	3,054	4.29
1953	92	180	3,030	111	399	2.18
1954	77	460	16,000	122	739	2.99
1955	65	260	2,520	131	690	2.74
1956	180	1,030	6,450	451	4,517	4.06
1957	428	610	4,570	238	2,134	3.79
1958	407	520	14,000	238	1,662	3.26
1959	358	250	5,000	148	636	2.43
1960	412	240	4,940	152	588	2.27
1961	598	190	2,370	122	441	2.18
1962	1088	220	4,420	139	490	2.15
1963	548	180	2,430	117	419	2.17
1964	340	210	2,650	131	514	2.30
1965	248	240	2,080	130	642	2.64
1966	175	450	2,860	212	1,450	3.22
1967	316	280	3,710	133	685	2.72
1968	247	430	4,450	201	1,274	3.07
1969	235	390	3,850	183	1,074	2.93
1970	323	310	3,500	170	884	2.72
1971	485	240	4,090	130	592	2.51
1972	778	240	4,340	134	586	2.45
1973	839	200	4,510	120	476	2.31
1974	535	200	4,220	123	480	2.28
1975	722	180	8,140	114	401	2.15
1976	918	160	4,090	98	325	2.07
1977	757	176	4,260	104	388	2.23
1978	540	212	4,580	106	462	2.45

Year	SNL-NM recorded gamma dose data			Lognormal fit		
	No. of workers reported gamma dose ≥ 50 mrem	Gamma dose (mrem)		Gamma dose (mrem)		GSD
		Mean	Maximum	Median	95%-ile	
1979	306	236	3,990	120	553	2.53
1980	252	186	3,070	111	447	2.33
1981	137	274	3,210	151	727	2.59
1982	231	380	4,340	178	1,090	3.01
1983	231	250	4,360	123	570	2.55
1984	116	268	3,560	144	688	2.59
1985	150	263	2980	138	688	3.97
1986	105	292	2630	153	863	4.42
1987	143	320	3,580	143	869	2.99
1988	188	232	2,260	139	602	2.44
1989	111	198	1,194	130	507	2.29
1990	88	205	1,497	120	523	2.45
1991	90	170	1,256	99	395	2.31
1992	43	206	924	147	545	2.21
1993	37	180	521	134	479	2.17
1994	36	192	828	129	508	2.30
1995	38	151	495	119	348	1.92
1996	33	250	844	167	747	2.49
1997	29	230	972	141	633	2.49
1998	36	194	945	115	512	2.48
1999	25	189	603	135	518	2.26
2000	30	189	720	141	498	2.16
2001	3	83	95	Use 141	-- <sup>a</sup>	-- <sup>a</sup>
2002	1	52	52	Use 141	-- <sup>a</sup>	-- <sup>a</sup>
2003	14	135	374	Use 141	-- <sup>a</sup>	-- <sup>a</sup>
2004	64	164	572	137	377	1.85
2005	36	128	396	112	253	1.64

a. Too few data are available from 1949, 2001, 2002, and 2003 to adequately define a distribution. It is recommended that the median value given (highest surrounding year) be used as an estimate of unmonitored doses for these years.

Table 6-5. SNL-NM worker nonpenetrating dose statistics, 1987 to 2005 (Widner 2006).

Year	SNL-NM recorded gamma dose data			Lognormal fit		
	No. of workers reported shallow dose ≥ 50 mrem	Nonpenetrating dose (mrem)		Nonpenetrating dose (mrem)		GSD
		Mean	Maximum	Median	95%-ile	
1977 <sup>a</sup>	190	117	1,000	93	249	1.82
1978 <sup>a</sup>	116	168	1,060	111	427	2.26
1979 <sup>a</sup>	88	195	1,580	130	477	2.20
1980 <sup>a</sup>	74	121	740	95	248	1.79
1981 <sup>a</sup>	23	127	750	99	272	1.85
1982 <sup>a</sup>	43 <sup>b</sup>	150	500	117	361	1.99
1983 <sup>a</sup>	38	186	870	127	491	2.28
1984 <sup>a</sup>	19	151	590	108	391	2.19
1985	36	272	2080	137	767	4.39
1986	28	176	530	130	463	2.98
1987	138	370	4,050	152	1,017	3.18
1988	125	310	2,260	165	908	2.82
1989	112	221	1,392	137	575	2.39
1990	105	236	1,691	138	628	2.51
1991	130	203	1,639	112	507	2.51
1992	46	356	3,565	175	914	2.73
1993	54	216	702	145	622	2.42
1994	39	236	1,034	157	648	2.37
1995	42	180	750	135	437	2.04
1996	45	306	1,225	179	991	2.83
1997	48	256	1,906	138	678	2.63
1998	53	201	1,217	119	513	2.44

Year	SNL-NM recorded gamma dose data			Lognormal fit		
	No. of workers reported shallow dose $\geq$ 50 mrem	Nonpenetrating dose (mrem)		Nonpenetrating dose (mrem)		GSD
		Mean	Maximum	Median	95%-ile	
1999	33	218	858	146	629	2.43
2000	34	226	902	159	637	2.32
2001	3	105	126	-- <sup>c</sup>	-- <sup>c</sup>	-- <sup>c</sup>
2002	1	206	206	-- <sup>c</sup>	-- <sup>c</sup>	-- <sup>c</sup>
2003	17	121	374	105	252	1.70
2004	85	164	611	131	404	1.99
2005	55	129	396	109	276	1.76

- For these years, values are based on those worker dose records in which both penetrating and nonpenetrating dose were 50 mrem or greater because those are the data that are available in a form that supports analysis at this time.
- One extreme value of 49.93 rem nonpenetrating dose in 1982 was excluded from this analysis as accident related, most certainly investigated in detail for the individual involved, and not typical of anticipated worker exposures.
- Too few data are available from 2001 and 2002 to adequately define a distribution. It is recommended that the median value from 2000 be used as an estimate of unmonitored doses for these years.

## 6.4 UNCERTAINTY FACTORS

For the usual analysis of measured film badge doses, it is possible to read a photon dose of 100 mrem to within  $\pm 15$  mrem if the exposure involved photons with energies between several keV and several MeV (Morgan 1961). The estimated standard error in recorded film badge doses from photons of any energy is  $\pm 30\%$ .

The situation for neutrons was not as favorable as for photons. With NTA films used at Sandia before 1971, the estimated standard error was likely larger and varied significantly with the energy of the neutrons. Measured neutron doses to workers were very likely underestimated. The recommended approach is to use neutron-to-photon dose ratios to estimate the neutron dose using the more reliably measured photon doses.

Given the lack of specific technical information obtained with respect to dosimetry systems for much of SNL-NM history, it is necessary to estimate respective measurement uncertainty based on reported values for contemporary systems in use at other facilities. Table 6-6 shows estimates based on the technical basis document for the Hanford Site (ORAUT 2004), with additional values for NTA film adapted from ORAUT 2006. Some general analogies can be drawn between Hanford and SNL-NM in terms of dosimetry technologies employed during various time periods of interest.

Table 6-6. Bias and uncertainty estimates (adopted from ORAUT 2004 and ORAUT 2006).

Dosimeter	Period of use	Bias magnitude and range		Uncertainty factors	
		Overall bias <sup>a</sup>	Range in bias	Systematic <sup>b</sup>	Random <sup>c</sup>
Two-element film	1949–1958	1.27	1.13–1.60	1.2	1.8
Multi-element film	1959–1971	1.02	0.86–1.12	1.1	1.4
Two-chip TLD	1971–1973	1.12	1.04–1.16	1.05	1.2
Multi-element TLD	1974–current	1.00	0.95–1.05	1.05	1.2
NTA film <sup>d</sup>	1959–1971	1.5	0.5–1.5	1.5	

- Divide recorded dose by bias value to determine deep dose
- Systematic uncertainty related to lack of knowledge concerning energy distributions and geometries
- Random uncertainty due to variation across workers, energy levels, and geometries
- The most favorable to claimant estimates from the various energy ranges in ORAUT 2006 are applied to all applicable energy ranges.

## 6.5 MISSED BETA/PHOTON DOSE

Missed dose is the dose that might not have been accounted for on an individual's records because of loss of or damage to an individual's dosimeter or because the records might have indicated zero dose due to the detection limitations of the film or TLD.

For the usual analysis of measured film badge doses, MDLs in the literature range from about 30 to 50 mrem for beta/photon irradiation (West 1993; Wilson et al. 1990). MDLs for the most recently used TLDs (Model 8801 and 8802) are estimated based on Walker (1997a), Stanley (1980), and Potter (1993b). Because of the lack of site specific data on MDLs for the early TLDs used 1971-1990, these dosimeters are estimated to have had MDLs between those for the film badges and the more advanced TLDs.

Several exchange frequencies were in use at any one time, so the dose reconstructor needs to determine from individual records which exchange frequency applies to a specific worker. The values in the last two columns of Table 6-7 can be considered maximum annual missed doses for the purpose of dose reconstructions. Beginning with 1979, badge exchange frequencies are assumed to be monthly if specific information to the contrary for the worker is not available (NIOSH 2002a).

Table 6-7. SNL beta/photon dosimeter period of use, type, minimum detection limit (MDL), exchange frequency, and potential annual missed doses.

Period of use	Dosimeter	Deep MDL <sup>a</sup> (mrem)	Nonpenetrating MDL <sup>a</sup> (mrad)	Exchange frequency	Geometric mean annual missed dose <sup>b</sup>	
					Deep dose (mrem)	Nonpenetrating dose (mrad)
1949 to ~1958	Film in metal holder with open window and Pb filter	40	Not measured	Biweekly (n=26) <sup>c</sup>	520	Not applicable
				Monthly (n=12)	240	
				Quarterly (n=4)	80	
1959 to April 1971	Film in plastic holder with open window, Al, Cd, and Sn filters.	40	40	Biweekly (n=26) <sup>c</sup>	520	520
				Monthly (n=12)	240	240
				Quarterly (n=4)	80	80
May 1971 to ~1973	2271 TLD	20	38	Monthly (n=12)	120	228
				Quarterly (n=4)	40	76
1973 to ~1990	Two-card neutron TLD	20	38	Monthly (n=12)	120	228
				Quarterly (n=4)	40	76
~1990 to ~1994	8801 TLD <sup>d</sup>	10	35	Monthly (n=12)	60	210
		10	35	Quarterly (n=4)	20	70
~1995 to present	8802 TLD <sup>d</sup>	5	5	Monthly (n=12)	30	30
		8	10	Quarterly (n=4)	16	20

- Estimated MDLs for each dosimeter technology in the workplace.
- Mean annual missed dose calculated using MDL/2 from NIOSH (2002a).
- Dosimeters in reactor areas (Kingsley 1971) and for organizations handling radioactive materials consistently were commonly exchanged on a biweekly basis (i.e., every two weeks; Kingsley 1953b).
- References for TLD MDLs: Walker (1997a), Stanley (1980), Potter (1993b).

Missed beta/photon dose is entered into Interactive RadioEpidemiological Program (IREP) as a lognormal distribution with a geometric mean consistent with Table 6-7 and a geometric standard deviation (GSD) of 1.52.

## 6.6 PARTITIONING OF BETA/PHOTON DOSES TO ENERGY CATEGORIES

Very little spectroscopy data to indicate gamma spectrum in SNL-NM work areas have been found. To estimate the gamma spectrum to which workers were exposed, facilities were grouped into categories.

Reactor facilities have dispersed fields of higher energy photons from fission as well as fission and activation products.

Radioactive materials handling and processing facilities included a wide variety of activities. For electron accelerator facilities, bremsstrahlung photons dominate the secondary radiation field. Thick shields of concrete or other materials result in photons in the MeV energy range. For proton or positive ion accelerators, neutrons generally constitute the greatest hazard. Thick shields designed for the neutron hazard eliminate all but the most energetic photons. With shielding and safety interlock systems, exposure to the direct beam of SNL-NM accelerators was rare and the personal exposure records should document them. However, maintenance personnel were exposed to activated accelerator components during repairs, target changes, etc. One reaction of particular importance is the thermal neutron capture of sodium in the concrete of the accelerator shielding (NCRP 2003). This reaction [ $^{23}\text{Na}(n,\gamma)^{24}\text{Na}$ ] produces a radioactive isotope that decays with a 15-hr half-life by emitting gamma rays of 1.4 and 2.8 MeV. Therefore, the gamma energy spectrum for accelerators was judged to consist primarily of photons with energies above 250 keV based on Argonne National Laboratory-East data (ORAUT 2006). Radiation from X-ray machines and some radioisotopes presented low-energy photon hazards, but these appear to have been generally used in conjunction with higher energy sources rather than in distinct facilities.

Table 6-8 lists the energy ranges for beta and photon exposures at SNL-NM. SNL-NM has possessed separated plutonium and sources of low-energy X-rays, so non penetrating doses could have resulted from exposures to low-energy photons as well as beta particles.

Table 6-8. Recommended beta and photon radiation energies and percentages for SNL-NM facilities.

Process/ buildings	Description	Operations		Radiation type	Energy selection, keV	Percentage
		Begin	End			
Plutonium component operations	Weapon component inspection, testing, and assembly:	1949	Present	Beta photon	>15 <30 30–250	100 65 <sup>a</sup> 35 <sup>a</sup>
Reactors (Area V)	<u>During Operation:</u> Highly dispersed fields of higher energy photon radiation fields from fission process, activation and fission product nuclides. Potentially narrow beams of higher energy neutron radiation from test ports, etc., into reactor core. Potential for significant airborne nuclides and there could be significant higher energy beta radiation.			Beta photon	>15 30–250 >250	100 25 75
	<u>Not in Operation:</u> Highly dispersed fields of higher energy photon radiation fields from activation and fission product nuclides. No significant neutron radiation. Possibly higher energy beta radiation during maintenance work resulting from fission products.					
	Various reactors (see Table 2-2 for listing and date ranges)		1958	Present		
Uranium component operations	Weapon component inspection, testing, and assembly: depleted and enriched uranium.	1949	Present	Beta photon	>15 30-250	100 100
Thorium component operations	Weapon component inspection, testing, and assembly:	1959?	Present	Beta photon	>15 30–250 >250	100 50 50
Accelerator operations (in Area IV)	Simulation of weapon environments, material and component testing, fusion and particle beam research:			Beta photon	>15 30-250 >250	100 10 90
	Various accelerators (see Table 2-2 for a listing and date ranges)		1958	Present		
ANT/nosetip payload operations	Exposure from Se-75, Ta-182, Ir-192 sources	1974	1980	Beta photon	>15 30–250 >250	100 30 70
Calibrations and irradiations	Onsite irradiation of instruments, dosimeters, components:			Beta photon	>15 30–250 >250	100 5 95
	Gamma Source Range (Cs-137)		1962?	Present?		
	Vertical Range (Co-60 and Ir-192)		1975?	Present?		
	GIF (Co-60, Cs-137)		1962	Present?		

Process/	Description	Operations		Radiation	Energy	Percentage
HCF	Isotope production of molybdenum-99 for medical use.	1998	2003?	Beta photon	>15 30–250 >250	100 40 60
Waste handling	Radiation characteristics highly dependent on source of waste.	1949	Present	Beta photon	>15 30–250 >250	100 50 50

a. Low-energy photons were not measured by early SNL-NM film badges that had no unfiltered areas. For years before 1959, 100% of the measured photon (gamma) dose should be attributed to the 30-250-keV category. An additional dose of 1.876 times the measured dose should be attributed to the low-energy photon category (<30-keV) based on Traub et al. (2005).

Plutonium facilities have mostly 17-keV photons with some 59-keV photons from <sup>241</sup>Am. Plutonium handled at SNL had been separated from fission products. Low-energy photons that would have been received by workers handling plutonium components were not measured by early SNL-NM film badges that had no unfiltered areas. Before 1959, when nonpenetrating radiation was first measured (and reported in terms of beta dose), 100% of the measured photon (gamma) dose should be attributed to the 30 to 250-keV category. An additional dose of 1.86 times the measured dose should be attributed to the low-energy photon category (<30 keV). This factor is based on calculated dose rates due to 30 to 2,560-keV photons, based on measured dose, (P2/C7) for a generic 6-kg pit (5 y), from Table A.3 of Traub et al. (2005).

From 1959 through 1966, nonpenetrating dose (reported as beta dose) should be attributed to the <30-keV category. From 1967 to 1986, nonpenetrating dose (reported as such, “NPR”) should be attributed to the <30-keV category. From 1987 to the present, shallow dose should be attributed to the <30-keV category.

## 6.7 UNMONITORED NEUTRON DOSE

There should not typically be significant neutron exposure of unmonitored workers. However, if estimation of neutron dose is called for, the recommended option is to apply the neutron-to-photon distribution data from Table 6-9 to measured or estimated missed or unmonitored photon doses. This process of calculating a neutron dose is based on the assumption that the distribution of neutron doses to unmonitored workers is equivalent to the distribution of neutron doses measured for monitored workers.

Table 6-9. Recommended distributions for neutron-to-photon ratio.

Neutron source type	Neutron-to-photon dose ratio		
	GM	GSD	95%-tile
General operations 1987 to present—(based on 16 annual records between 1987 and 2005 <sup>a</sup> )	0.36	2.55	1.70
General operations in earlier years—(based on 38 annual records from 1977 through 1984 <sup>a</sup> )	0.39	4.97	5.43

a. Reference: Widner 2006

The distributions of neutron-to-photon ratios should not be used to estimate neutron doses when neutron measurements are available after 1971, when thermoluminescent dosimeters replaced NTA film at SNL-NM for neutron dosimetry.

The application of the data from Table 6-9 to measured or estimated photon doses can be accomplished through Monte Carlo simulation. Alternatively, to obtain an overestimate of unmonitored neutron dose, the appropriate 95th-percentile value from Table 6-9 can be applied to the measured or estimated photon dose for each period of interest. The “general operations” values from Table 6-9 are based on the reported annual doses for 1987-2005 and for 1977 through 1984 for individuals who had *both* gamma and neutron dose results of 100 mrem or greater (Widner 2006).

This assessment was based on 16 cases for 1987-2005 and 38 cases for 1977 through 1984. Annual dose data obtained for 1987 through 2005 in electronic form include “facility type” codes. It might be possible with additional work to break down neutron-to-photon ratios by facility type, but the number of occurrences of annual dose records with both gamma and neutron doses high enough to yield meaningful ratios might be very low and limit the usefulness of that approach.

## 6.8 MISSED NEUTRON DOSE

Neutron radiation has been present at SNL in association with various weapons components, reactors, and accelerators. For other locations, missed neutron dose is very unlikely because of the very low potential for neutron exposure. To calculate the missed neutron dose, the dose reconstructor must first determine if the person worked near neutrons and the category of neutrons. This can best be determined by examining the work location records and whether a worker or others in the badge reporting group were assigned any neutron dose equivalent. If no neutron dose was assigned to the worker or coworkers for several months, the dose reconstructor should assume that the person was not exposed to neutrons.

For periods before 1971, distribution of neutron-to-photon ratios from Table 6-9 should be applied to the recorded gamma doses to estimate missed neutron doses. Table 6-10 lists the neutron missed dose for those exposed to neutrons.

Table 6-10. Neutron dosimeter period of use, type, MDL, exchange frequency, and potential annual missed dose.

Period of use	Dosimeter	MDL (mrem)	Exchange frequency	Mean annual missed dose (mrem) <sup>a</sup>
1959 to April 1971	Film Badge 2 with NTA film	<50	Monthly (n=12)	<300 <sup>b</sup>
May 1971 to 1989	First TL dosimeter and Model 2271 TLD badge <sup>c</sup>	30	Monthly (n=12)	180
			Quarterly (n = 4)	60
1990 to present	Model 8801/8802 TLD badges <sup>c</sup>	10	Monthly (n=12)	60
			Quarterly (n = 4)	20

a. Mean annual missed neutron dose calculated using MDL/2 from NIOSH (2002a).

b. Neutron-to-photon ratio should be used to estimate missed doses during these periods.

c. Reference for 8801/8802 badge MDLs is Walker (1997a). The LLD for neutron doses on the 8801 dosimeter is a function of neutron dose and ranges from 0.010 to 0.120 rem for low energy to 14 MeV neutrons. Because of the lack of site specific data on MDLs for the early TLDs used 1971-1989, these dosimeters are estimated to have had MDLs between those for NTA film and the more advanced TLDs.

## 6.9 PARTITIONING OF NEUTRON DOSES TO ENERGY CATEGORIES

Table 6-11 lists default neutron dose fractions by energy range for SNL operational areas where neutron exposures were possible along with the associated ICRP 60 correction factors (ICRP 1991). The neutron dose equivalent is calculated by multiplying the recorded neutron dose by the area-specific correction factors.

Because of the wide variety of reactors used at SNL-NM, and paucity of neutron spectrum data for them, an assumption of 100% fission spectrum neutrons (0.1 to 1 MeV) is used. The energy distribution for plutonium component handling is based on data from the Los Alamos National Laboratory TDD (ORAUT 2005d). Energy characteristics for neutron generators is based on Ward (1993). Data for PuBe sources are based on Buckner and Sims (1992). Neutrons from Cockroft-Walton generators vary based on target and energy applied, but are most commonly 2.5 MeV and 14.1 MeV (Elliott 1971). The energy breakdown for other accelerators is based on a combination of the data for the Zero Gradient Synchrotron and Intense Pulsed Neutron Source from the TBD for Argonne National Laboratory-East (ORAUT 2006).

Table 6-11. Recommended dose fractions and ICRP 60 correction factors for SNL neutron sources.

Process	Facility description	Operations		Neutron energy	Default dose fraction (%)	ICRP 60 correction factor
		Begin	End			
Plutonium component operations	Weapon component inspection, testing, and assembly:					
	Plutonium handled at SNL had been separated from fission products.	1949	Present	<10-100 keV	11	0.23
				0.1-2 MeV	56	1.1
2-20 MeV	33	0.44				
Reactor operations	Neutrons of varying energies from reactor operations:					
	Various reactors (see Table 2-2 for a listing and date ranges)	1958	Present	<10-100 keV	0	—
				0.1-2 MeV	100	1.9
2-20 MeV				0	—	
Neutron generators	Testing of neutron sources for weapons applications (typically 14 MeV).	1959	1997+	<10-100 keV	0	—
				0.1-2 MeV	0	—
				2-20 MeV	100	1.3
Calibration and irradiation	Onsite irradiation of instruments, dosimeters, components:					
	Neutron source range	1950?	Present	<10-100 keV	0	—
	PuBe neutron sources	1950?	Present	0.1-2 MeV	20	0.38
2-20 MeV				80	1.1	
Cockroft-Walton Generator	Simulation of weapon environments, material and component testing:					
	Sandia C-W accelerator facility	Late 1950s	Present	<10-100 keV	5	0.11
				0.1-2 MeV	5	0.095
2-20 MeV				90	1.2	
Other accelerators	Various accelerators (see Table 2-2 for a listing and date ranges)	1958	Present	<10-100 keV	5	0.11
				0.1-2 MeV	14	0.27
				2-20 MeV	46	0.61
				>20 MeV	35	0.35

The following sources of photon/beta radiation identified in Table 6-8 are not listed in Table 6-11 because they, as conducted at SNL-NM, are not significant sources of neutron exposure: uranium component operations, thorium component operations, ANT/Nosetip payload operations, hot cell facility (HCF) operations, and routine waste handling. Neutron generator testing is not a significant source of photon/beta exposure, so it is listed only in Table 6-11 below. Cockroft-Walton generator operations, a component of Accelerator Operations included in Table 6-8, are listed below as a potentially significant source of neutron exposures.

## 6.10 RECOMMENDED DOSE CONVERSION FACTORS

Dose conversion factors (DCFs) should be selected from NIOSH (2002a) for the dose quantities specified in Table 6-12 for the periods of interest.

Table 6-12. Recommended DCFs for SNL dose assessments.

Period	Recommended photon DCFs	Recommended neutron DCFs
1949–1971	Exposure ( $R$ ) to organ dose equiv. ( $H_T$ )	Deep dose equiv. ( $H_{p,slab}(10)$ ) to organ dose equiv. ( $H_T$ )
1972 to present	Deep dose equiv. ( $H_p(10)$ ) to organ dose equiv. ( $H_T$ )	

## 7.0 ATTRIBUTIONS AND ANNOTATIONS

Where appropriate in the preceding text, bracketed callouts have been inserted to indicate information, conclusions, and recommendations to assist in the process of worker dose reconstruction. These callouts are listed in this section with information that identifies the source and justification for each item. Conventional references are provided in the next section that link data, quotations, and other information to documents available for review on the Oak Ridge Associated Universities (ORAU) Team servers.

- [1] Buddenbaum, Jack E., Certified Health Physicist (CHP). ENSR. Health Physicist. 2007. Statements about name changes to the site are based on discussions with Harold Rarrick in February 2006 that were confirmed with Tracy Ikenberry on February 9, 2007.
- [2] Kathren, Ronald L., CHP, The Kathren Group, Inc., 2005. This summary statement in the TBD is taken from discussions during the development of this document in 2005 with SNL-NM retirees and current staff about the types of preemployment X-ray examinations and equipment used at SNL-NM.
- [3] Kathren, Ronald L., CHP, The Kathren Group, Inc., 2005. A review of SNL employee X-ray records was performed to derive this information
- [4] Kathren, Ronald L., CHP, The Kathren Group, Inc., 2005. This summary statement in the TBD is taken from discussions during the development of this document in 2005 with SNL-NM retirees and current staff about the X-ray examination protocol and how it changed over the years.
- [5] Argall, Rowena S., CHP. ENSR. Health Physicist. 2007. A summary of the history of SNL from information in previous sections of the Site Profile is included here for completeness.
- [6] Argall, Rowena S., CHP. ENSR. Health Physicist. 2007. This information is included to alert the dose reconstructor that additional dosimetry records might need to be requested if there is indication that the worker could have visited other SNL sites during the employment period and there is no indication of dosimetry records in the file or requests to those facilities. This statement is included here for completeness.
- [7] Argall, Rowena S., CHP. ENSR. Health Physicist. 2007. This statement is taken from documented discussions during 2005 through February 2007 with SNL retiree H. Rarrick about his activities at nuclear storage sites and SNL general activities at storage sites and from information in previous sections of the Site Profile.
- [8] Argall, Rowena S., CHP. ENSR. Health Physicist. 2007. A summary of the activities of SNL from information in previous sections of the Site Profile is included here for completeness.
- [9] Argall, Rowena S., CHP. ENSR. Health Physicist. 2007. This summary statement is taken from discussions during 2005 through February 2007 with SNL retirees and current staff about the types of activities at SNL and the perceived intake potentials from those activities.

- [10] Argall, Rowena S., CHP. ENSR. Health Physicist. 2007.  
The discussion in this section is included to assist the dose reconstructor to determine the potential for exposure for a worker based on activities that might have been described as "assembly" of weapons. The information is from the cited document.
- [11] Argall, Rowena S., CHP. ENSR. Health Physicist. 2007.  
The nuclides of primary concern are based on those nuclides listed in documents and reports and discussions with SNL retirees and current staff.
- [12] Argall, Rowena S., CHP. ENSR. Health Physicist. 2007.  
Tables 5-1 and 5-2 are provided based on the cited reference and information in other sections of the document. These locations are summarized here for completeness and to provide the dose reconstructor with a source of information on the intake potential of various areas on the site.
- [13] Argall, Rowena S., CHP. ENSR. Health Physicist. 2007.  
The table of nuclear weapons tests is included here to assist the dose reconstructor in recognizing the names of these tests in the event that these names are mentioned in the telephone interview or other records. While these tests are included in other documents, they are included here for completeness.
- [14] Argall, Rowena S., CHP. ENSR. Health Physicist. 2007.  
This statement is based on review of various documents that are cited throughout Section 5.0 and conversations with SNL retirees and current staff between 2005 and February 2007.
- [15] Argall, Rowena S., CHP. ENSR. Health Physicist. 2007.  
This paragraph is based on review of various documents that are cited throughout Section 5.0 and conversations with SNL retirees H. Rarrick and D. Thompson and current SNL personnel between 2005 and February 2007. The retirees participated in the bioassay programs. Mr. Rarrick was involved in the analysis of the samples, and Mr. Thompson was involved in the archiving of the data. Research is currently being done to retrieve the bioassay results that were archived during that period. Because of statements made during interviews with individuals currently involved with the bioassay program, out-sourced bioassay and a Memorandum of Understanding with LANL were considered to be the prime source of bioassay results for years before 1991. However, information from Mr. Rarrick and Mr. Thompson is contradictory to the outsourcing of *in vitro* samples. Because the methods of that time were similar to those used at LANL for uranium and tritium, MDAs listed for LANL are most likely applicable to the analyses at SNL when no other information is available.
- [16] Argall, Rowena S., CHP. ENSR. Health Physicist. 2007.  
The compromise of the integrity of the results was not immediately apparent to SNL. When SNL realized that the results were not valid, resampling was not therefore a practical option for many of the workers. According to Potter (1994a) the total numbers of samples, excluding spikes, were 531 baseline, 96 termination, 28 routine, and 46 incident samples. Resampling took place for 10 of the individuals approximately 2 years after the initial sample. However, because fission products, tritium, and uranium bioassay were involved, the validity of resampling to assess intakes for the previous period should be carefully assessed during the dose reconstruction.

- [17] Argall, Rowena S., CHP. ENSR. Health Physicist. 2007.  
The reference cited lists a number of sites that reported positive results in control subjects counted with the Helgeson system during that period. Because the majority of counted workers in 1989 and 1990 at SNL also had positive results, Sam Glover, NIOSH Health Physicist, suggested inclusion of this reference in the document. However, it is favorable to the claimant to include the positive results in the dose reconstruction.
- [18] Argall, Rowena S., CHP. ENSR. Health Physicist. 2007.  
This statement is based on review of worker dosimetry records and conversations with SNL retirees and current staff between 2005 and February 2007.
- [19] Argall, Rowena S., CHP. ENSR. Health Physicist. 2007.  
This statement is based on review of worker dosimetry records and conversations with SNL retirees and current staff between 2005 and February 2007.
- [20] Argall, Rowena S., CHP. ENSR. Health Physicist. 2007.  
Conversations with SNL retirees between 2005 and February 2007 indicated that bioassay was being conducted and analyzed on site as early as 1949. The program was not formalized because there was a perception that there was little need for a routine bioassay program. Less than 100 workers participated annually in bioassay in the pre-1992 era. This is contrary to the perception of current SNL personnel. Conversations with current personnel indicated that bioassay was extremely rare and, if it was conducted, the samples were sent off site.
- [21] Argall, Rowena S., CHP. ENSR. Health Physicist. 2007.  
This statement is based on conversations with SNL retiree D. Thompson in February 2007. Mr. Thompson participated in the archiving of the dosimetry records beginning in 1967.
- [22] Argall, Rowena S., CHP. ENSR. Health Physicist. 2007.  
This statement is based on review of records supplied by LLNL.
- [23] Argall, Rowena S., CHP. ENSR. Health Physicist. 2007.  
This statement is based on conversations with active SNL personnel between 2005 and February 2007.
- [24] Argall, Rowena S., CHP. ENSR. Health Physicist. 2007.  
This statement is included in the discussion to show that SNL reviewed the requirements for bioassay and made decisions accordingly.
- [25] Argall, Rowena S., CHP. ENSR. Health Physicist. 2007.  
The bioassay results that have been reviewed have been accompanied by background and standard calibration results. According to conversations with retirees, the procedures used to analyze these samples were standard protocol similar to the protocol used by LANL. Therefore, the MDAs reported by LANL are generally applicable.
- [26] Argall, Rowena S., CHP. ENSR. Health Physicist. 2007.  
Although the actual procedures used by the Industrial Hygiene department have not been found at this time, conversations in February 2007 with retirees indicated that the analytical protocol for analysis of bioassay was basically the same as that used at LANL during the time periods. Therefore, the assumption of MDAs in the same range can be made when no other information is available.

- [27] Argall, Rowena S., CHP. ENSR. Health Physicist. 2007.  
Review of the radionuclides that are associated with the locations on the site and at offsite locations and conversations with SNL retirees and current staff indicated that plutonium was not a consistent source term across the site.
- [28] Argall, Rowena S., CHP. ENSR. Health Physicist. 2007.  
This statement is based on review of various documents that are cited throughout Section 5.0 and conversations with SNL retirees and current staff between 2005 and February 2007.
- [29] Argall, Rowena S., CHP. ENSR. Health Physicist. 2007.  
This statement is based on conversations in February 2007 with SNL retiree H. Rarrick.
- [30] Argall, Rowena S., CHP. ENSR. Health Physicist. 2007.  
This statement is based on conversations in February 2007 with SNL retiree H. Rarrick.
- [31] Argall, Rowena S., CHP. ENSR. Health Physicist. 2007.  
This statement is based on review of various documents that are cited throughout Section 5.0 and conversations with SNL retirees and current staff between 2005 and February 2007.
- [32] Argall, Rowena S., CHP. ENSR. Health Physicist. 2007.  
This statement is based on review of various documents that are cited throughout Section 5.0 and conversations with SNL retiree H. Rarrick and current SNL personnel between 2005 and February 2007.
- [33] Argall, Rowena S., CHP. ENSR. Health Physicist. 2007.  
This statement is based on review of various documents that are cited throughout Section 5.0 and conversations with SNL retirees and current staff between 2005 and February 2007.
- [34] Argall, Rowena S., CHP. ENSR. Health Physicist. 2007.  
Until alpha spectroscopy was available to analyze the electroplated samples, only gross alpha was counted either on a proportional counter or NTA film. However, the label on the results might have been  $^{239}\text{Pu}$ . Analysis involving alpha spectroscopy would include results for both  $^{239}\text{Pu}$  and  $^{238}\text{Pu}$ . It is not possible to distinguish between  $^{239}\text{Pu}$  and  $^{240}\text{Pu}$  using alpha spectroscopy techniques, therefore the activity of  $^{240}\text{Pu}$  is included in the activity listed for  $^{239}\text{Pu}$ .
- [35] Argall, Rowena S., CHP. ENSR. Health Physicist. 2007.  
The term *pure*  $^{238}\text{Pu}$  was typically used in reference to a mixture of  $^{238}\text{Pu}$  that is actually *predominately*  $^{238}\text{Pu}$ , as shown by the ratios of the mixtures listed in the cited references. The pure  $^{238}\text{Pu}$  mixture is found in heat source technology work.
- [36] Bihl, Donald. Hanford. Health Physicist. 2006.  
Calculations were performed to determine ratios using the Pu.ex software (Rittman 1984).
- [37] Argall, Rowena S., CHP. ENSR. Health Physicist. 2007.  
This statement is based on review of various documents that are cited throughout Section 5.0 and conversations with SNL retirees and current staff between 2005 and February 2007.
- [38] Argall, Rowena S., CHP. ENSR. Health Physicist. 2007.  
This statement is based on a conversation with C. A. Potter, CHP, manager of the Internal Dosimetry Department in 2006.

- [39] Argall, Rowena S., CHP. ENSR. Health Physicist. 2007.  
This statement is based on review of various documents that are cited throughout Section 5.0 and conversations with SNL retirees and current staff between 2005 and February 2007.
- [40] Argall, Rowena S., CHP. ENSR. Health Physicist. 2007.  
Records of tritium surveys with positive results are found throughout the archives for various locations on the site, including the locations given as examples in this section.
- [41] Argall, Rowena S., CHP. ENSR. Health Physicist. 2007.  
Records of tritium surveys with positive results are found throughout the archives for various locations on the site, including the locations given as examples in this section.
- [42] Argall, Rowena S., CHP. ENSR. Health Physicist. 2007.  
This protocol was established for LLNL tritium doses as an overestimate.
- [43] Argall, Rowena S., CHP. ENSR. Health Physicist. 2007.  
This statement is based on review of various documents that are cited throughout Section 5.0 and conversations with SNL retirees and current staff between 2005 and February 2007. While the general internal dosimetry program was based on the concept that the majority, if not all, of the airborne radioactivity comprised nonrespirable particles, the examples of operations in this paragraph indicated that there were situations that could produce respirable particles of uranium and other radionuclides.
- [44] Argall, Rowena S., CHP. ENSR. Health Physicist. 2007.  
This statement is from the Internal Dosimetry Program procedures. Special bioassay sampling would only be initiated by a routine sample that exceeded the limits when a routine sampling program was actually in place.
- [45] Argall, Rowena S., CHP. ENSR. Health Physicist. 2007.  
This statement is based on conversations with C. A. Potter in 2006.
- [46] Argall, Rowena S., CHP. ENSR. Health Physicist. 2007.  
This statement is based on conversations with C. A. Potter in 2006.
- [47] Argall, Rowena S., CHP. ENSR. Health Physicist. 2007.  
This statement is based on conversations with C. A. Potter in 2006 and SNL retirees D. Thompson and H. Rarrick in 2007.
- [48] Argall, Rowena S., CHP. ENSR. Health Physicist. 2007.  
This statement is based on conversations with SNL retiree H. Rarrick in 2006 and 2007. Mr. Rarrick stated that the protocol and analytical technique in the early days were based in general on standard techniques in use in the industry at the time. Protocol similar to that used at LANL was used at SNL, therefore the analytical sensitivities can be considered to be similar.
- [49] Argall, Rowena S., CHP. ENSR. Health Physicist. 2007.  
This decision to classify an intake as occupational based on whether the concentration was within the expected range from dietary uranium intakes is known to be the practice since 1993. Whether any similar decisions to classify positive results were made before that time is not known.

- [50] Argall, Rowena S., CHP. ENSR. Health Physicist. 2007.  
Conversations with SNL retirees D. Thompson and H. Rarrick indicated that uranium bioassay was performed and recorded at SNL as early as 1949.
- [51] Argall, Rowena S., CHP. ENSR. Health Physicist. 2007.  
This statement is included as an example of a contamination incident and the SNL response to such incidents, in which engineering controls were used prevent a reoccurrence. This type of response was frequently indicated in documents that were reviewed.
- [52] Argall, Rowena S., CHP. ENSR. Health Physicist. 2007.  
This statement is based on the review of numerous log sheets that contain air sample results. These log sheets span many years of the history of SNL.
- [53] Argall, Rowena S., CHP. ENSR. Health Physicist. 2007.  
This statement is based on conversations with C. A. Potter in 2006.
- [54] Argall, Rowena S., CHP. ENSR. Health Physicist. 2007.  
This statement is based on conversations with C. A. Potter in 2006.
- [55] Argall, Rowena S., CHP. ENSR. Health Physicist. 2007.  
This statement is based on conversations with SNL retiree H. Rarrick in 2006.
- [56] Argall, Rowena S., CHP. ENSR. Health Physicist. 2007.  
This statement is based on conversations with C. A. Potter in 2006.

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## GLOSSARY

### activation

The induction of radioactivity in material by irradiation with neutrons.

### activity fraction

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

### aging

In the context of reactor fuel and mixtures of plutonium isotopes, the time since <sup>241</sup>Am was separated from the plutonium mixture.

### alpha radiation

Positively charged particle emitted from the nuclei of some radioactive elements. An alpha particle consists of two neutrons and two protons (a helium nucleus) and has an electrostatic charge of +2. Alpha particles have low penetrating power and a short range (a few centimeters in air). Outside the body, the most energetic alpha particle generally fails to penetrate the dead layers of cells covering the skin or a sheet of paper. Alpha particles emitted by radionuclides inside the body are a more significant health risk.

### becquerel (Bq)

International System unit of radioactivity equal to 1 disintegration per second; 1 curie equals 37 billion ( $3.7 \times 10^{10}$ ) Bq.

### beta radiation

Charged particle emitted from some radioactive elements with a mass equal to 1/1,837 that of a proton. A negatively charged beta particle is identical to an electron. A positively charged beta particle is a positron. Most of the direct fission products are (negative) beta emitters. Exposure to large amounts of beta radiation from external sources can cause skin burns (erythema), and beta emitters can be harmful inside the body. Thin sheets of metal or plastic can stop beta particles.

### dosimetry

Measurement and calculation of internal and external radiation doses.

### exposure

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

### fabrication

Manufacturing.

### implosion

A sudden inward compression and reduction in volume.

### Kiva

A remotely controlled critical assembly building.

### rad

Unit of absorbed dose.

**radioactivity**

Disintegration of certain elements (e.g., radium, actinium, uranium, and thorium) accompanied by the emission of alpha, beta, gamma, and/or neutron radiation from unstable nuclei.

**rem**

Traditional unit of radiation dose equivalent that indicates the biological damage caused by radiation equivalent to that caused by 1 rad of high-penetration X-rays multiplied by a quality factor. The average American receives 360 millirem a year from background radiation. The sievert is the International System unit; 1 rem equals 0.01 sievert. The word derives from roentgen equivalent in man; rem is also the plural.

**simulated**

In the context of urine sampling, collection of urine from about one-half hour before retiring to bed, through the sleep period, and for about one-half hour after rising for two consecutive nights, or other similar protocol, to simulate a 24-hour sample.

**Sv**

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

**thermoluminescent dosimeter (TLD)**

Device for measuring radiation dose that consists of a holder containing solid chips of material that, when heated by radiation, release the stored energy as light. The measurement of this light provides a measurement of absorbed dose.

**tolerance values**

Refers to the concentration of a radionuclide in a bioassay sample, above which would indicate that an unacceptable intake had occurred or an unacceptable body burden existed in that individual.