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Dose Reconstruction
Project for NIOSH

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ACRONYMS AND ABBREVIATIONS

AEC	U.S. Atomic Energy Commission
ALARA	as low as reasonably achievable
AP	anterior-posterior
Bq	becquerel
CEDE	committed effective dose equivalent
CEP	Controls for Environmental Pollution
CFR	Code of Federal Regulations
Ci	curie
cm	centimeter
cpm	counts per minute
CRDL	Chemical and Radiation Detection Laboratory
DAC	derived air concentration
DCF	dose conversion factor
DOE	U.S. Department of Energy
DOELAP	DOE Laboratory Accreditation Program
dpm	disintegrations per minute
DU	depleted uranium
EEOICPA	Energy Employees Occupational Illness Compensation Program Act of 2000
ESE	entrance skin exposure
g	gram
GPS	Gas Purification System
gU	generic unit
hr	hour
HTO	tritiated water vapor
HVAH	high-velocity air hood
HVL	half-value layer
ICP-MS	inductively coupled plasma mass spectrometry
ICRP	International Commission on Radiological Protection
ICT	Insulating Core Transformer (LLNL)
in.	inch
IREP	Interactive RadioEpidemiological Program
keV	kiloelectron-volt, 1,000 electron-volts
kVp	peak kilovoltage, applied kilovoltage
L	liter
LAT	lateral
LLD	lower limit of detection
LLNL	Lawrence Livermore National Laboratory
m	meter
MBA	Mass Balance Area

mCi	millicurie
MDA	minimum detectable activity
MDC	minimum detectable concentration
MDL	minimum detection limit
MeV	megaelectron-volt, 1 million electron-volts
mL	milliliter
mrem	millirem
MPC_a	maximum permissible concentration in air
nC	nanocoulomb
nCi	nanocurie
NIOSH	National Institute for Occupational Safety and Health
NIST	National Institute of Standards and Technology
PA	posterior-anterior
pCi	picocurie
POC	probability of causation
PRDP	Personnel Radiation Dosimetry Program
R	roentgen
RDC	Radiation Detection Company
RESL	Radiological and Environmental Sciences Laboratory
SID	source-to-image distance
SNL/CA	Sandia National Laboratories/California
SNL/NM	Sandia National Laboratories/New Mexico
SOP	Safe Operating Procedure
SPD	site profile document
SSD	source-to-skin distance
SWP	Safe Work Permit
T_2	elemental tritium
TBD	technical basis document
TIB	technical information bulletin
TLD	thermoluminescent dosimeter
TMA/EAL	Thermo-Analytical Incorporated/EAL Corporation
TRL	Tritium Research Laboratory
U-nat	natural uranium
VERS	Vacuum Effluent Recovery System
yr	year
µCi	microcurie
µg	microgram
§	section or sections

1.0 INTRODUCTION

1.1 PURPOSE

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 at 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[1]] 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

[1] The U.S. Department of Labor is ultimately responsible under the EEOICPA for determining the POC.

1.2 SCOPE

This SPD for the Sandia National Laboratories site in Livermore, California (SNL/CA) describes aspects of the SNL/CA site and historical activities and practices that are pertinent to dose reconstruction under the EEOICPA. Section 2.0 describes the site and the activities. Section 3.0 addresses the protocol and procedures associated with routine occupational medical X-rays of SNL/CA employees. Estimated occupational intakes associated with above-background ambient levels of radionuclides in the SNL/CA environment are provided in Section 4.0, along with estimated external ambient dose rates. Sections 5.0 and 6.0 address the technical issues associated with measurement of internal and external dose, respectively.

Attributions and annotations, indicated by bracketed callouts and used to identify the source, justification, or clarification of the associated information, are presented in Section 7.0.

2.0 SITE DESCRIPTION

2.1 PURPOSE AND SCOPE

This section briefly describes the physical environment of the SNL/CA site and the site activities and processes carried out since its establishment in 1956. The types of radioactive materials present on the SNL/CA site, the areas in which exposure to radioactive materials may have occurred, and the emissions of radionuclides to the environment are identified. Access control for radioactive areas is also described.

2.2 INTRODUCTION

The background information in this introductory section was taken primarily from Ullrich (2003), which was prepared to support DOE's compliance with the national Historic Preservation Act. The SNL/CA site was established in 1956 to provide direct support for Lawrence Livermore National Laboratory (LLNL) nuclear weapons designs. The primary mission during the time coinciding with the Cold War (1956 to 1989) was the design and testing of non-nuclear components of nuclear weapons designed by LLNL. SNL/CA was to engineer, or "weaponize," the nuclear physics packages designed by LLNL; production of parts and final weapons was accomplished at other weapons complex sites.

Sandia Corporation, a Lockheed Martin Company, currently operates SNL/CA and Sandia National Laboratories/New Mexico (SNL/NM) in Albuquerque. From 1956 to 1993, SNL/CA was managed and operated by American Telephone and Telegraph. In 1993, the contract was awarded to Martin Marietta Corporation, now known as Lockheed Martin Corporation (DOE 2003).

The SNL/CA site presently consists of approximately 70 buildings/facilities on 410 acres and is located just across East Avenue from the LLNL in Livermore, California. Figure 2-1 shows the SNL/CA facilities in relation to the boundaries; Figure 2-2 shows the individual buildings/facilities on the site.

The SNL/CA site initially consisted of a long narrow strip of 50 acres stretching south from East Avenue. A personnel building was the first building to be completed in September 1957; the rest of the original buildings (warehouse, model shop, environmental test, and central steam plant, office and laboratory building) were completed in 1958. Design support for LLNL was originally provided by a small group of engineers and support staff. Initially, 14 SNL/NM employees worked with LLNL in LLNL facilities. In 1957, SNL/CA began using the LLNL Site 300 for explosive testing. By 1958, with over 800 employees, SNL/CA worked on the W38 warhead for Titan I and Atlas missiles. Support for LLNL also expanded to include effects test analyses and telemetry for LLNL nuclear tests of nuclear weapon designs. As part of the Plowshare Program, which ended in the early 1970s, SNL/CA also moved into evaluation of nuclear detonation in 1959 (Ullrich 2003).

The environmental test building for the new SNL/CA site (completed by the end of 1958) was used for some testing of new designs. During the 1960s, more support facilities were built, including storage and a maintenance shop. Additional test facilities (the centrifuge and Explosive Test Facility) were also constructed. In 1970, 86 acres were added to the site, providing an additional buffer area.

Although SNL/CA has never lost their core mission of nuclear ordnance design and testing, the laboratory moved further into scientific research in the late 1960s, bringing in scientists, mathematicians, and materials specialists to work in applied research. SNL/CA's initial move into research on ^{3}H grew out of its familiarity with, and use of, ^{3}H in components. In 1974, the first

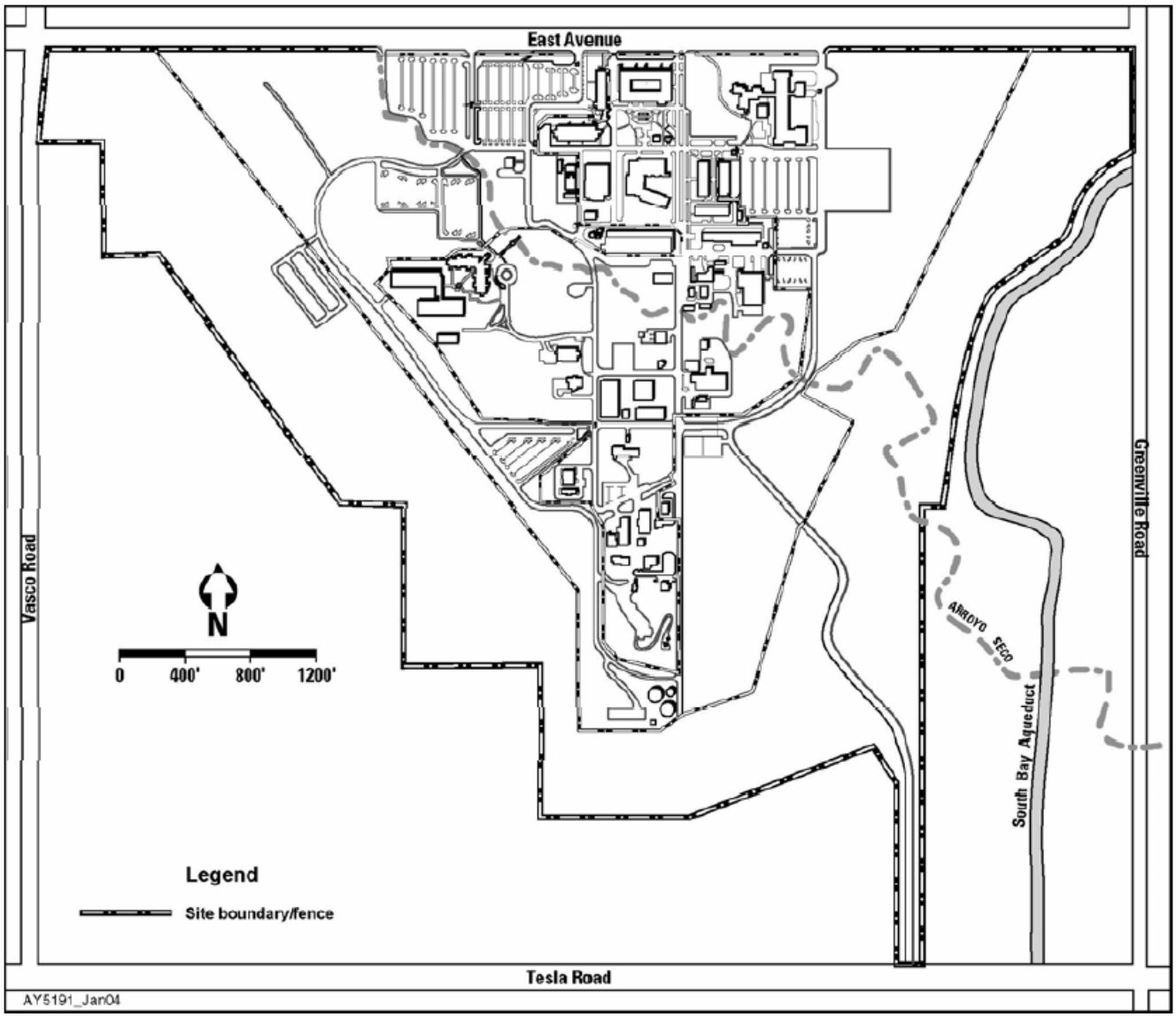

Figure 2-1. Boundaries of SNL/CA property (SNL/CA 2005).

structure dedicated to ^{3}H research at SNL/CA, the Tritium Research Laboratory (TRL), was added to the site, with completion of the basic laboratory building during the summer of 1975 (Garcia and Gorman 1996). The TRL became operational in late 1978.

Ranging further from weapons engineering, SNL/CA also pursued combustion research and, during the 1973 to 1974 energy crisis, began conducting some research into alternative energy. The expanded purpose of SNL/CA resulted in an increased variety of facilities within the site. In addition to the TRL, a large complex for the Combustion Research Facility was completed in 1980. An additional 24 acres of land was added as a buffer zone on the east side near the Tritium Facility in 1979. In 1986, an additional 228 acres was obtained, allowing an alternative exit route from the facility. Finally, in 1998, SNL/CA took part in a small land exchange to create a consistent buffer zone line along the western boundary, in which 2.82 acres were received in exchange for 5.41 acres. This brought the site's area down to 410 acres where it remains today. In 2004, there were 1094 employees at the site (SNL/CA 2005).

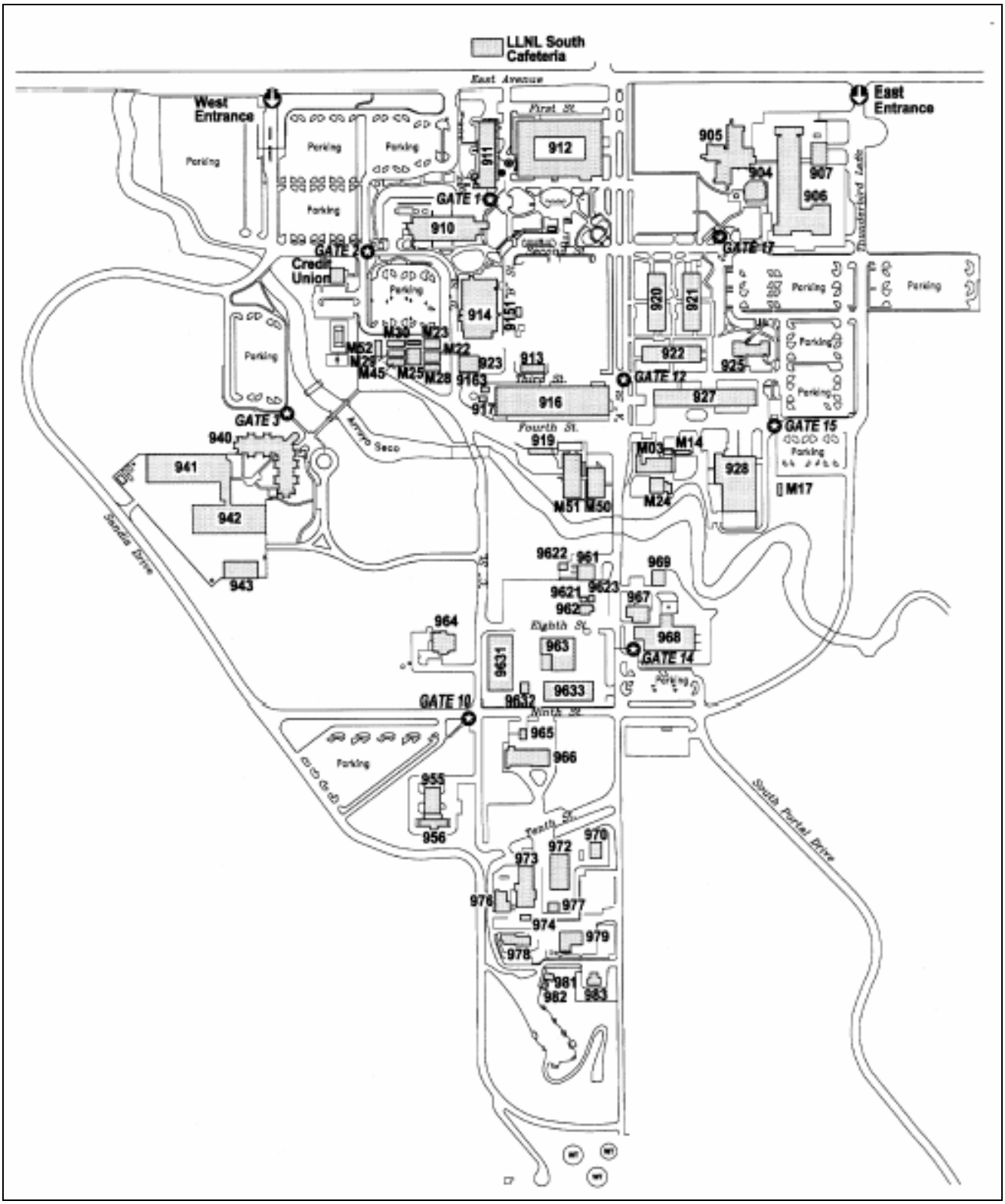

Figure 2-2. Map of SNL/CA facilities (Ullrich 2003).

2.3 SITE ACTIVITIES

In support of the various missions identified in Section 2.2, a number of activities were undertaken at SNL/CA over the years, some of which involved handling and release of radioactive materials. A complete list of buildings along with known information about present and past uses and the presence of radioactive materials are given in Attachment A. A subset of this building list can be alternatively compiled to represent the major process complexes at the SNL/CA site that handled radioactivity in some manner over the years (as shown in Table 2-1) and the predominant radiologically related activities that took place in these complexes.

Table 2-1. Area information and parameters.

Area	**Weapons Laboratory Facility Complex**
Description	Building 910, 912, 913, 914, 916, 918
Time Period	1958–1998
Activities	Test/repair neutron detectors, wet machining of DU, radiography of weapons components, radiography for materials science studies, ^{3}H storage studies, ion beam analysis of materials, Radiflo leak tests
Radionuclides	DU, ^{3}H, neutron generator, small accelerators, small sealed sources, small amounts of ^{85}Kr

Area	**Radiography**
Description	Building 923
Time Period	unknown–Early 1990s
Activities	radiography using X-rays, gamma rays, neutrons, alpha and beta particles
Radionuclides	^{60}Co, ^{192}Ir, ^{252}Cf, X-ray machines

Area	**Micro and Nano Technologies Laboratories**
Description	Buildings 941, 942, and 943
Time Period	unknown–present
Activities	radiography for materials science studies
Radionuclides	In Building 941 only: X-ray, ^{238}U, and beta sources (sealed)

Area	**Former Tritium Research Laboratories (Currently the Chemical and Radiological Detection Laboratory)**
Description	Buildings 967, 968, and 969
Time Period	1974–1996
Activities	^{3}H research: >0.1 g ^{3}H handled in glovebox, 0.0005–0.1 g in high velocity air hoods
Radionuclides	^{3}H, DU (no radionuclides after decommissioning complete in 1996)

Area	**Explosives and Environmental Testing Complex**
Description	Buildings 955, 956, 966, 972, 974, 976, 977, 978, 979, 981, 983
Time Period	1958–present
Activities	environmental testing of mock-up weapons and components
Radionuclides	DU

Area	**Storage Facilities**
Description	Buildings 921, 927, 961, 982
Time Period	unknown–present
Activities	storage and packaging of waste materials
Radionuclides	^{3}H, DU, natural Th, Trace ^{239}Pu and mixed fission products

According to SNL/CA annual environmental reports dating back to 1983, the laboratory typically handled kilogram amounts of depleted uranium (DU), gram amounts of ^{3}H (when the TRL was

operational), and only microcurie quantities of other isotopes (SNLL 1983a). However, the 1992 Environmental Impact Statement for LLNL and SNL/CA noted 100 Ci ^{192}Ir and ^{60}Co radiographic isotopes and many other smaller sealed sources with activity ranging from 1 µCi to 500 mCi, stored in a shielded radiography cell (DOE 1992).

DU is, and has been, largely in the form of alloyed metal components, often encapsulated. Wet machining of uranium metal did occur over the years (Adolphson 1972, SNL/CA 1989a, Wallace 1988). Powdered DU sealed in storage containers for ^{3}H-storage studies (as the tritide) has also been present in approximately 1-kg amounts (SNL/CA 1991a). A classified activity involving powdered uranium hydride in gram amounts was conducted in gloveboxes in buildings 979, 916, and the TRL (Wright 2006). There were some thorium metal parts used on test systems as well (Wright 2006).

Tritium was generally handled in the TRL in the form of a gas, although effluents could be in the form of a gas, liquid, or solid. Mixed solid and liquid tritiated waste was generated; the majority of which was in the form of scintillation cocktails, which were shipped offsite for incineration in Florida (DOE 1992). Details regarding total quantities of ^{3}H present for each year of operation of the TRL and pertinent to types of research activities carried out in this facility are described in Section 2.3.1 below.

Most of the remaining radioactive materials were contained in sealed sources. However, a small amount of ^{85}Kr was used in the Radiflo leak detection studies. The period of use of the Radiflo units is not known.

In 1982, the *Final Environmental Impact Statement* for LLNL and SNL/CA (DOE 1982) indicated that the major activities taking place on the SNL/CA site were related to “tritium research; arming, fusing and firing systems; and aerodynamic and structural elements used in U. S. nuclear bombs and warheads.” At that time, it was found that operations within most facilities on site have no significant environmental impact and they handle radionuclides in small enough quantities such that special containment features and operating procedure are not required to assure that no radioactivity is released. The only exceptions to this finding of no potential significant impact were the SNL/CA radioactivity storage vaults, one of which was located in Building 927, and the other small vault within the TRL (Building 968), and the TRL itself. It was noted that the contents of the 927 vault varied according to work needs, but generally included depleted uranium-containing mock-ups of weapons, and small, sealed radioactive sources not in use. Tritium was not stored in the 927 vault; rather the small quantities present on site were kept in the laboratories where they were being used in Building 968, or in the vault in that building. Other than the H-3 source due to vaults in Building 968 (the TRL), there were experiments conducted with up to 120 g (1.2×10^{6} Ci) of ^{3}H per experiment in sealed gloveboxes.

In 1992, the *Final Environmental Impact Statement and Environmental Impact Report for Continued Operation of Lawrence Livermore National Laboratory and Sandia National Laboratories* (DOE 1992) concluded that the only building for which a potentially significant radiological release could occur under accident conditions would be Building 968, which housed the TRL at that time. This was based on a review of all potential radiological sources and the likelihood that the source could become dispersed into the onsite or offsite environment.

The 2003 *Final Site-wide Environmental Assessment of the Sandia National Laboratories/California* (DOE 2003) addressed, in part, the potential environmental impact of continuing operations at the SNL/CA. The TRL operations were discontinued in 1996 (including the decontamination activities); thus, the radiological activities at this time involved radioactive material management of legacy

radioactive material inventories and current nuclear material inventories, including radioactive isotopes used in laboratory research and radiation monitoring activities.
A Waste Management Site Plan from 1980 states:

> *The radwastes comprise, primarily, tritium and depleted uranium (D-38) but no Transuranics, fission products or induced activity. Now that the Tritium Research Laboratory is operational, tritium contamination will account for more than 50% of the radwaste volume in FY80. Depleted uranium contaminated wastes will account for most of the remainder. D-38 wastes are mostly machine turnings from the Machine Shop in Bldg. 913 and components from the Test Assembly Group also in Bldg. 913. Contaminated paper and machine lubricants make up most of the actual D-38 waste volume. D-38 components may be classified or unclassified*. (Wright 1981a)

The information provided above suggest that activities involving DU and ^{3}H are the only two potentially-significant sources of internal exposure to workers based largely on the fact that these are the only two radioactive materials present in significant quantities throughout the history of SNL/CA. External exposures to radiation from radiation-generating devices (small accelerators, radiography sources, neutron generators, X-ray machines) are also of concern. The remainder of Section 2.3 will be devoted to aspects of the SNL/CA site pertinent to these sources of exposure.

2.3.1 Tritium Research Laboratory

The TRL was designed as a modern research and development facility to provide support to the DOE weapons complex (Garcia and Gorman 1996). The TRL was the first major ^{3}H research and development complex to use secondary containment coupled with a cleanup system as a means to control personnel exposure to levels as low as reasonably achievable (ALARA) and reduce environmental releases (Garcia and Gorman 1996). The TRL continued to operate until 1993 when all tritium and associated research was transferred to other DOE facilities. The TRL was decontaminated from January 1994 through October 1996 and now houses the Chemical and Radiation Detection Laboratory (CRDL).

Administrative controls limited the amount of tritium in the research laboratory to a maximum of 120 g at any time (excluding the vault) and to a maximum of 300 g at the facility in total at any time (Wall 1981). Table 2-2 lists the historical inventory of elemental tritium (T_2) at TRL (from Garcia and Gorman 1996).

The TRL was divided into 9 laboratories, 2 office areas, a ^{3}H storage room, a control room, a shop, a ^{3}H gas purification equipment room, and 8 miscellaneous rooms. The Radioactive Materials Area was separated from the rest of the facility by two sets of double doors. The ventilation system directed airflow from clean areas to areas of increasing contamination potential (Wright 1981a).

Operations in the TRL depended upon the types of research ongoing at the facility but were focused on the physical and chemical characterization of ^{3}H and its compounds, the fabrication of tritium compounds for use as engineering components (Wright 1981a), and examining the behavior of hydrogen isotopes and helium in metals to understand transport and structural properties (Garcia and Gorman 1996). Operations were performed inside sealed gloveboxes for any experiments involving more than 0.1 g ^{3}H (DOE 1992).

Building 968 was not operated as a security exclusion area. Building access was controlled by a sign in/sign out procedure until a computer-based security system was put in place, which is known to have been in place in 1991 (SNL/CA 1991b). Radiological materials were also dispensed through

one authorized person and amounts were administratively controlled. The Nuclear Material Management Group had three Mass Balance Areas (MBAs) and a custodian for each area was

Table 2-2. Historical T_2 inventory at TRL.

Year	T_2 (grams)	T_2 (Bq)
1979	1.06	3.8E+14
1980	15.47	5.5E+15
1981	29.62	1.1E+16
1982	28.55	1.0E+16
1983	27.62	9.8E+15
1984	37.86	1.3E+16
1985	85.49	3.0E+16
1986	107.58	3.8E+16
1987	174.35	6.2E+16
1988	181.59	6.4E+16
1989	148.89	5.3E+16
1990	115.46	4.1E+16
1991	117.85	4.2E+16
1992	131.67	4.7E+16
1993	101.21	3.6E+16
1994	29.43	1.0E+16
1995	9.96	3.5E+15
1996	4.04	1.4E+15
1997	0	0.0E+00

responsible for issuing of nuclear materials to personnel in his MBA. Material was accounted for to the nearest 0.01 gram, with amounts equal to or greater than 0.005 g rounded to the nearest 0.01 g.

Two central decontamination systems were used: (1) Gas Purification System (GPS) and (2) Vacuum Effluent Recovery System (VERS). The GPS was used to remove tritium, tritiated water, and tritiated hydrocarbons from the sealed glovebox atmosphere in the event of either a significant release or a slow buildup of background contamination. The VERS was used to remove tritium, tritiated water vapor (HTO), and, tritiated hydrocarbons from the glovebox pressure control system and the gases exhausted from all of the vacuum pumps in the laboratory before venting this gas to the stack.

2.3.2 Depleted Uranium Machining

Machine shops in Buildings 913, 914, and in an annex of Building 918 have been used for machining of DU fairly infrequently at SNL/CA. A 1972 memorandum from Adolphson (1972) indicates that “machining of uranium alloy test specimens” began in 1971. No machining has been done since 1998, when Building 913 was decommissioned. According to the Health Physicist at SNL/CA, this machining was done as a wet process; with the exception of one accident he was aware of where dry cutting occurred (Wright 2006). The 1989 Safe Operating Procedure (SOP) governing machining of uranium metal (SNL/CA 1989a) dictates that all machining operations shall be performed wet and that dry operations are prohibited. According to a 1985 DOE Headquarters appraisal of SNL/CA and response to findings (Wallace 1988), uranium machining occurs approximately once or twice a quarter. The appraisal/response report also indicates that the Hazards Control Division at SNL/CA, as of 1987, had 16 yr of air sampling data for this procedure, which “demonstrate that airborne contamination is not a problem during typical machining operations.”

2.3.3 Radiography, Accelerators, and Neutron Generators

Until the early 1990s, the Radiography Facility was a 3880 sq ft building (Building 923) which included X-ray machines, gamma ray source, neutron sources, and many alpha and beta sources (DOE 1992). The radiation sources were primarily used for the radioscopy and electron imaging of weapon and non-weapon components. The radiation sources included the radiation-producing machines with energies up to 420,000 volts; the 100 Ci ^{192}Ir and ^{60}Co radiographic isotopes; sealed ^{252}Cf spontaneous fission neutron source located inside a massive neutron source shield; and many other smaller sealed neutron, alpha, beta, and gamma ray isotope sources of low activity (one µCi to 500 mCi), which are stored in a shielded radiography cell. The building contained four shielded radiography cells. The sources were used infrequently for material characterization studies involving technologies such as radiation transmission gauging, backscatter measurements, X-ray fluorescence studies, and neutron activation analysis (DOE 1992).

X-Ray diffraction equipment is reported as being present in building 913, room 115, as early as 1965 (SNL/CA 1964). An incident that occurred with a diffractometer (see Section 2.4.1) was reported to have occurred in building 913, room 113 in 1979 (Lovell 1980). Radiography is currently conducted in Building 941, part of the Micro and Nanotechnologies Laboratory (DOE 2003).

Two small accelerators have been, and remain, located in Building 916 (DOE 1992, 2003). These include a 1 MeV Tandem accelerator and a 700/200 keV positive ion accelerator (SNLL 1983b, SNL/CA 1991c, Wright 2006).

Neutron generators used in the design, assembly, testing, calibration, and repair of neutron detectors at SNL/CA are located in Building 910 (SNL/CA 1990a). Neutron generator tests are also conducted in Building 974 (Attachment A).

2.4 SITE PROCESSES

2.4.1 Incidents

According to Garcia and Gorman (1996), a few incidents occurred with regard to the TRL which may have lead to ^{3}H exposures in excess of routine operations. Other sources (Lovell 1980, SNL/CA 1960-1983) listed excess exposure to ionizing radiation, although only one was attributable to a work-related exposure. These incidents are listed below:

Personnel Exposure to Ionizing Radiation, December 1979

During set-up of a high-temperature diffractometer attachment to port 2 of the X-ray unit tube tower, a worker was accidentally exposed to X-rays from port 1. This resulted in a dose assignment of 26 to 34 rem to the basal cell layer of a skin area (less than 400 cm^2) of the chest and arm (Lovell 1980).

Personnel Exposure and Tritium Release, July 1984
During disassembly of an engineering experiment in Laboratory 115A, an environmental release of approximately 2.5 Ci of HTO occurred. Seven employees received doses ranging from a high of 1,650 mrem down to 2 mrem.

Tritium Release, January 1986
A weld crack in a storage container caused a leak and release in the Decontamination Laboratory 115A causing an environmental release of 200 Ci of HTO. There were no resulting personnel exposures.

Personnel Exposure and Tritium Release, August 1987
An 1100 Ci environmental release occurred when an operator disassembled a vessel outside a glovebox in Laboratory 115. This release of elemental tritium gas (T_2) caused a building evacuation and the operator received a dose of 15 mrem.

Tritium Release, October 1988
An environmental release occurred during a GPS Regeneration operation in Laboratory 115A. The release was estimated at 124 Ci of HTO and no personnel exposure was involved.

Personnel Exposure and Tritium Release, March 1989
A personnel exposure of ~180 mrem occurred during VERS pump maintenance operations in Laboratory 120. An environmental release of approximately 11.5 Ci of HTO occurred. Subsequent operation required the flushing of pumps before pump maintenance operations.

Personnel Exposure and Tritium Release, October 1993
Six personnel received doses from 2 mrem to 5 mrem when a VERS pump failed and caused a tritium release in Laboratory 115A. An environmental release of 2 Ci HTO was reported by Garcia and Gorman (1996), but was alternatively reported as a 30 Ci stack release by Garcia (994a). The personnel doses came from both cleanup and pump replacement operations.

2.4.2 Effluents

The only radionuclides released due to normal operations from SNL/CA are small amounts of tritium and trace amounts of DU (SNLL 1982, 1983a, 1984; SNL/CA 1986 to 1988, 1989b, 1990b, 1991d, 1992, 1993, 1994a, 1995 to 2005). The DU effluents are reported to be less than 10 µCi ([3.7×10^{-5} Bq] for both liquid and airborne effluents collectively [SNLL 1982, 1983a, 1984; SNL/CA 1986]). In 1990, DOE (1990) reported that emissions of radioactive particulates from SNL/CA activities that potentially generate such particulates are controlled with high-efficiency particulate air (HEPA) filters. A total of 33 HEPA filters, installed in four buildings (913, 916, 961, and 979) were in use at that time. Tritium from the TRL has been the only airborne effluent routinely reported, and is often declared to be the only detectable effluent (SNL/CA 1986 to 1988, 1989b, 1990b, 1991d, 1992, 1993, 1994a, 1995 to 1997). Monitoring of tritium effluents from the TRL ceased in 1996 after the facility was transitioned for other uses (SNL/CA 1998).

Table 2-3 lists the measured effluents during the operational and decontamination periods of the TRL (Garcia and Gorman 1996). These quantities include the incidental releases noted in Section 2.4.1 above.

2.5 RADIOLOGICAL PROTECTION AND ACCESS CONTROL

Badging and bioassay programs were carried out throughout the history of operations at SNL/CA to provide information on exposure to workers. This information was used to limit annual exposures to workers to within exposure guidelines set forth by the U.S. Atomic Energy Commission (AEC), Energy Research and Development Administration, and DOE criteria (Kingsley 1968; SNL/CA 1994b, SNL/CA –undated a). When workers traveled to other locations, such as across the street to the LLNL site, the LLNL Site 300, or other non-SNL-operated sites, the host sites were to be requested to report exposures at those locations to the appropriate staff at SNL/CA or SNL/NM (SNL/CA - undated b). A response to the DOE site appraisal finding related to offsite dosimeters indicated that SNL/CA directed employees to “request internal and/or dosimetry, workplace monitoring, etc., if they must enter radiation controlled areas at the visited facility” in 1985 (Wallace 1988).

Dosimetry records for SNL/CA employees were transferred to the SNL/NM site in the late 1980s to the early 1990s (Hallman 1989, 1990; Perez-Romo 1994). In a 1994 memo (Perez-Romo 1994), it is indicated that dosimetry data on computer tapes from Reynolds Electric and Engineering Company (the contractor providing base support at the Tonopah Test Range where SNL/Tonopah employees worked) were also sent for storage at SNL/NM. It does not appear that the pre-1989 records for SNL/CA were incorporated into the currently accessible electronic database at SNL/NM for SNL employees. Dosimetry data covering the period 1973 – 1987 was recently recovered from SNL/NM, and data for 1988 was recovered from SNL/CA. Cumulative summary data covering the period prior to 1973 has also been recovered from SNL/CA. However, annual data for all years prior to 1973 is currently being sought.

All SNL/CA workers were required to wear dosimetry badges between 1959 and 1969, according to a collection of annual Summary of Whole-Body Radiation Exposures to ionizing radiation reports (SNL/CA 1958-1978). In 1958, the radiation exposure summary report indicates that 41 employees (of 537) were not monitored. However, a 1960 internal memo (SNL/CA 1958-1961) indicates that all employees were monitored from the time of establishment of the Sandia Livermore Corporation. It was noted in 1984 (Lovell 1984a, b) that Building 923 and Room 100 of Building 916 had sources and/or equipment capable of producing radiation levels that could exceed 500 mrem/yr (which was 10% of the Radiation Protection Standards for workers at that time). Thus, it was stated (Lovell 1984a,b) that radiation dosimeters would be required for access to these buildings during radiographic procedures and during accelerator operations. In 1965, DeSelm (1965) wrote a memorandum indicating that Buildings 911 (Medical), 913, 914, 916, 921, 9143, and Areas 8 and 9 could produce levels exceeding AEC' Manual Chapter 0524 (AEC 1963) criteria of 6 mR/hr for radiation exposure to the whole body or critical organs for wearing dosimeters.

Table 2-3. Effluents of ^{3}H to air and sewer during operations and cleanup of TRL.

Year	Total tritium in stack discharges to air (Bq)	Estimated HTO to air (Bq)	Total tritium (as HTO) in wastewater discharges to sewer (Bq)
1979	2.2E+11	Not estimated	1.5E+08
1980	9.3E+11	Not estimated	1.5E+08
1981	1.6E+12	Not estimated	5.7E+09
1982	7.5E+12	Not estimated	1.1E+10
1983	3.5E+12	2.7E+12	1.4E+10
1984	6.1E+12	5.4E+12	1.4E+10
1985	1.9E+13	1.4E+13	7.4E+10
1986	2.7E+13	2.3E+13	9.3E+08
1987	6.8E+13	2.1E+13	8.7E+09
1988	5.8E+13	3.8E+13	1.7E+10
1989	3.1E+13	2.4E+13	1.1E+10
1990	1.1E+13	9.0E+12	7.4E+09
1991	1.7E+13	1.3E+13	4.6E+09
1992	9.8E+12	5.0E+12	2.3E+09
1993	7.0E+12	4.9E+12	2.5E+09
1994	3.5E+12	3.4E+12	2.2E+09
1995	2.7E+12	2.7E+12	8.9E+08
1996	2.9E+09	Not estimated	0.0E+00

Other than requiring dosimeters for access to certain facilities at SNL/CA, there are other forms of access control in place. The 700/200 keV accelerator (Room 104, Building 916) and 1 MeV tandem accelerator were fitted with interlocks to prevent entry during operation (SNL/CA 1991c, SNLL 1983b). Access was also restricted to the machine shop (Room 119) in Building 913 during uranium

machining operations by roping off the area and having supervisors present during all such operations (SNL/CA 1989a). For the TRL (Building 968), building access was controlled by a sign in/ sign out procedure until a computer-based security system was put in place, as noted in Section 2.3.1.

3.0 OCCUPATIONAL MEDICAL DOSE

As part of the requirements for employment at SNL/CA starting in 1956, some employees received periodic physical examinations. These could include annual radiographic examinations of the chest, as well as lumbar spine X-rays and lateral (LAT) chest X-rays at hire. Because these examinations were required for employment, the *External Dose Reconstruction Implementation Guidelines* (NIOSH 2002) and 42 C.F.R. pt. 82, require the X-ray doses to be part of the occupational radiation exposure. This section of this SPD discusses medical screening X-rays required as a condition of employment; it does not include diagnostic and therapeutic exposures that were not required for employment.

The following sections describe the methodology used to estimate absorbed dose from X-ray exposure for SNL/CA workers. Section 3.1 describes X-ray examination frequency at SNL/CA as reconstructed from claimant files. Section 3.2 provides information on equipment and techniques used at SNL/CA, including assumptions necessitated by lack of protocol, measurement, or records data. Section 3.3 provides organ dose estimates by calendar year and type of X-ray. Section 3.4 documents uncertainties.

3.1 EXAMINATION FREQUENCY

Protocol for the frequency and type of X-ray examinations for SNL/CA workers from 1956 through the present time has not been located. A protocol for the frequency of chest X-ray examinations as a function of job category has also not been located and equipment type is not known. However, claimant files available at the time of this documents preparation generally indicated that a single posterior-anterior (PA) chest X-ray examination was performed at hire, annually, and possibly at termination from 1956 through the 1980s. They also showed that anterior-posterior (AP) and LAT lumbar spine X-rays were at least sometimes performed at hire as late as 1971. LAT chest X-rays were rarely taken; only two files (of over 50 files available at the time) indicated that LAT chest X-rays had been taken at hire. LAT chest X-rays should not be included in the default estimates for SNL/CA employees unless it is noted in their medical files that they were taken.

According to the health physics department, worker X-rays ceased in the 1980s, but no specific date is known. The X-ray machine equipment was removed from the site in approximately 1990 (Wright 2006). No evidence of the use of photofluorography has been found. It is recommended that annual chest X-rays be assumed through 1989 until documentation of the date of cessation of X-rays is found.

3.2 EQUIPMENT AND TECHNIQUES

The analysis assumed that radiological practices followed standards of medical practice to minimize dose to the worker; however, the type of equipment, technique factors, and some machine calibrations are not known. Medical records did record the beam current (mA), applied kilovoltage (kVp), and distance (presumably from machine to image) for PA chest examinations and occasionally for lumbar spine X-rays. The notation of mA in many of the records is likely the exposure (mAs), as the setting recorded in claim files (5 mA) is too low to have produced chest radiographs [1]. Additionally, the distance is assumed to be the source to image distance (SID) as it does not vary in the worker files reviewed (it is consistently 72 inches). However, exposure time, filtration, and entrance skin exposure (ESE) were not noted, nor was the use of screens or grids if applicable. A medical X-ray unit was present on the site; personnel report that the X-ray machine was removed in about 1990 and was not replaced. In the 1950s, the pre-employment set of X-rays may have been performed at SNL/NM or off the site in California.

This section of the SPD provides organ dose estimates from occupational X-ray examinations administered at SNL/CA from 1956 through 1969, 1970 to 1985, and post-1985 using calculated site-specific estimates with input from *Technical Information Bulletin* [TIB]*: Dose Reconstruction from Occupationally Related Diagnostic X-ray Procedures* (ORAUT 2005a). For the years before 1970, the default values from ORAUT 2005a assume minimal beam collimation and a half-value layer (HVL) of 2.5 mm Al. For 1970 to 1985, the default values assume that the beams were collimated and the HVL was 2.5 mm Al. For post-1985, the default values assume collimation and an HVL of 4.0 mm Al. These HVLs were used in the site-specific calculations as site-specific information was not available.

For all periods, the analysis should assume that a single PA chest X-ray occurred at hire, at each annual physical examination, and at termination of employment, as evidenced by available medical records. AP and LAT lumbar spine X-rays were taken once, at hire, for the years 1956 to 1971 [2]. Most workers are assumed to have received these X-rays. LAT chest X-rays were rarely taken, but have been included here for those energy employees with LAT chest X-rays (taken at hire or at a regular physical examination) noted in their medical files (two claim files with this X-ray view have been found). Dose reconstructors should assign dose from the X-ray procedures listed on the pre-employment X-ray record form. If the claim file is missing the pre-employment X-ray record, the dose reconstructor should assign dose from a PA chest and AP and LAT lumbar spine as the default for the pre-employment X-rays.

Efforts will continue to locate related SNL/CA X-ray protocol information. Until that is located, information from claim files related to SNL/CA have been reviewed and notations regarding machine settings (specifically mAs, kVp, and SID) have been used to calculate site-specific estimates of dose received from occupationally required X-ray examinations. The X-ray machine has been assumed to be single phase [3].

3.3 ORGAN DOSE ESTIMATES

This section discusses organ dose estimates. Section 3.3.1 describes the methodology used to estimate these doses and Section 3.3.2 discusses results.

3.3.1 Parameters and Estimation Method

ICRP (1982) guidance uses the following parameters to estimate air kerma and absorbed dose:

- Source to image distance (SID) in centimeters (cm)
- Total filtration (millimeters of aluminum, mmAl)
- Estimate of tissue thickness (AP and LAT)
- Machine settings (mA, exposure time [seconds], kVp, film size, and machine type)

If measured air kerma are available, these should be used. For SNL/CA, air kerma was estimated from Figure 3-1 (ICRP 1982) assuming a single-phase machine was used at SNL/CA. Assumptions (from medical records within claim files) were used to estimate air kerma for chest X-rays for pre-1970, 1970 to 1985, and post-1985 time periods. For the pre-1970 and 1970 to 1985 periods, 80 kVp, 5 mAs, a total filtration of 2.5 mmAl, and a source to subject distance (SSD) of 153 cm results in an air kerma estimate of 0.06 mGy per mAs at 1 meter. Total filtration of 2.5 mmAl is a reasonable estimate for those time periods (ORAUT 2005a). The entrance kerma is then calculated to be 0.13 mGy, as shown in Table 3-1. The LAT chest entrance kerma is 2.5 times the PA chest entrance kerma (ORAUT 2005a).

For post-1985, 110 kVp, 300 mA and 1/30 second exposure time, and SID of 183 cm (from claim files), with a total filtration estimate of 2.5 mmAl and HVL of 4.0 (ORAUT 2005a) result in an estimated air kerma of 0.10 mGy/mAs and calculated entrance kerma of 0.43 mGy for PA chest X-rays (Table 3-1). The LAT chest X-ray entrance kerma is 2.5 times the PA chest entrance kerma (ORAUT 2005a).

For lumbar spine X-rays, an applied kilovoltage of 75 kVp and beam current of 75 mAs were assumed for AP lumbar spine X-rays and 85 kVp and 150 mAs for LAT lumber spine X-rays based on information in claim files. A total filtration of 2.5 mmAl and HVL of 2.5 mmAl were assumed (ORAUT 2005a), resulting in an estimated air kerma of 0.045 mGy/mAs for AP lumbar spine X-rays and 0.070 mGy/mAs for LAT lumbar spine X-rays. A SSD of 63 cm for AP and 52 cm for LAT X-rays was used, resulting in entrance kermas of 8.5 mGy and 38.8 mGy for AP and LAT X-rays, respectively.

Entrance kerma for PA and LAT chest X-rays and for AP and LAT lumbar spine X-rays are presented in Table 3-1. Dose conversion factors (DCFs) are listed in Table 3-2.

Table 3-1. Entrance kerma by procedure and period.

Period	PA chest entrance kerma (cGy)	LAT chest entrance kerma, (cGY)	AP lumbar spine entrance kerma (cGY)	LAT lumbar spine entrance kerma (cGy)
Pre-1970	0.013[a]	0.0325[a]	0.85[b]	3.9[c]
1970–1985[a]	0.013	0.0325	NA	NA
Post-1985[d]	0.043	0.1075	NA	NA

a. Based on 80 kVp and 5 mAs, as observed in SNL/CA claim files and 2.5 mmAl total filtration (ORAUT 2005a). LAT chest entrance kerma is 2.5 times PA chest entrance kerma. PA entrance kerma = (0.06 mGy/mAs) x (5 mAs) x $(100cm/153cm)^2$ = 0.13 mGy.
b. Assumes 75 kVp and 75 mAs, as observed in SNL/CA claim files, and total filtration of 2.5 mmAl (ORAUT 2005a). Entrance kerma = (0.045 mGy/mAs) x (75 mAs) x $(100cm/63cm)^2$ = 8.5 mGy.
c. Assumes 85 kVp and 150 mAs, as observed in SNL/CA claim files, and total filtration of 2.5 mmAl (ORAUT 2005a). Entrance kerma = (0.070 mGy/mAs) x (150 mAs) x $(100cm/52cm)^2$ = 38.8 mGy.
d. Based on 110 kVp, 300 mA and 1/30 second exposure time (noted in one claim file) and total filtration of 2.5 mmAl (ORAUT 2005a). LAT chest entrance kerma is 2.5 times PA chest entrance kerma. PA entrance kerma = (0.10 mGy/mAs) x (10 mAs) x $(100cm/153cm)^2$ = 0.43 mGy.

The International Commission on Radiological Protection (ICRP) tables used to estimate absorbed dose (ICRP 1982) do not include all the organs included in the Interactive RadioEpidemiological Program (IREP) computer program. For organs in IREP but not identified in the ICRP tables, the dose conversion coefficient that is anatomically closest to the IREP-specified organs can usually be used to estimate dose. For example, the factor for lung can be applied to all other organs in the thoracic cavity, such as the esophagus and bone surface. For abdominal organs (bladder, colon), the

Table 3-2. DCFs (mGy per Gy air kerma); absorbed dose (1 mGy) for organs at various Al HVL for radiography (ORAUT 2005a; ICRP 1982).[a]

	Lumbar spine DCFs		Chest DCFs					
	Pre-1970 (2.5-mmAl HVL)	Pre-1970 (2.5-mmAl HVL)	Pre-1970 (2.5-mm Al HVL)		1970–1985 (2.5-mm Al HVL)		Post-1985 (4.0-mm Al HVL)	
Organ	**LAT**	**AP**	**LAT**	**PA**	**LAT**	**PA**	**LAT**	**PA**
Thyroid	0.01	0.3	137	174[b]	115	32	164	78
Eye/brain	0.01	0.3	137	32	115	32	164	78
Ovaries	N/A[c]	N/A[c]	N/A	N/A	0.6	1	2.5	5.2
Liver/gall bladder/spleen	14[d]	79[d]	220	451	220	451	351	674
Urinary bladder	N/A[d]	N/A[d]	N/A	N/A	0.6	1	2.5	5.2
Colon/rectum	N/A[d]	N/A[d]	N/A	N/A	0.6	1	2.5	5.2
Testes	N/A[c]	N/A[c]	N/A	N/A	0.1	0.01	0.1	0.01
Lungs (male)	14	79	193	419	193	419	313	628
Lungs (female)	14	79	220	451	220	451	351	674
Thymus[d]	14	79	220	451	220	451	351	674
Esophagus[d]	14	79	220	451	220	451	351	674
Stomach[d]	14	79	220	451	220	451	351	674
Bone surfaces[d]	14	79	220	451	220	451	351	674
Remainder[d]	14	79	220	451	220	451	351	674
Female breast	13[e]	25[e]	255	49	255	49	343	116
Uterus	31	287	N/A	N/A	0.6	1.3	2.1	5.2
Bone marrow (male)	22	37	37	92	37	92	76	178
Bone marrow (female)	22	37	29	86	29	86	59	172
Skin[f]	1.35[f]	1.35[f]	1.35[f]	1.35[f]	1.35[f]	1.35[f]	1.40[f]	1.40[f]

a. DCFs for lumbar spine are from ICRP 1982 and DCFs for chest are from ORAUT 2005a.
b. Per ORAUT (2005a), DCF for AP cervical spine corrected for depth by 0.2.
c. N/A = not applicable; organ dose values for the testes and ovaries for lumbar spine reflect actual measurements reported in Lincoln and Gupton (1958).
d. Using analogs listed in Table 3-4.
e. DCFs for lumbar spine examination not given in ICRP (1982). Values for the respective upper gastrointestinal examinations were used instead.
f. Backscatter factor from NCRP 102 (1989, Table B-3) (ORAUT 2005a); see Table 3-5.

dose coefficient for ovaries is used. This approach should be either favorable to the claimant or neutral. Table 3-3 lists analogs for IREP organs, as originally presented in ORAUT (2005a).

Table 3-3. Analogs for IREP organs not specified in ICRP (1982).

Anatomical location	ICRP #34 reference organ	IREP organ analogs[a]
Thoracic cavity	Lung	Thymus, esophagus, stomach, bone surface, liver/ gall bladder, remainder organs
Abdominal cavity	Ovaries	Urinary bladder, colon/rectum
Head and neck	Thyroid	Eye/brain

a. ORAUT 2005a

3.3.2 Organ Dose Estimates

Table 3-4 lists calculated organ dose estimates from PA and LAT chest X-ray examinations for each period (pre-1970, 1970 to 1985, and post-1985), although LAT chest X-rays were discontinued after 1970. The estimates for exposure from chest X-rays for these periods have been calculated using information taken from claim files in conjunction with assumptions from ORAUT 2005a.

Table 3-5 presents dose from AP and LAT lumbar spine X-rays that should be used for years 1956 to 1971. These X-rays were often taken at hire.

Table 3-4. Organ dose estimates for chest X-rays (rem).

Organ	Pre-1970 estimated dose[a,b,c] HVL = 2.5 mm Al (uncollimated)		1970–1985 estimated dose[a,b,c] HVL = 2.5 mm Al (collimated)		Post-1985 estimated dose[a,b] HVL = 4.0 mm Al (collimated)	
	LAT	PA	LAT	PA	LAT	PA
Thyroid	4.38E-03	2.26E-03	3.68E-03	4.16E-04	1.75E-02	3.35E-03
Eye/brain	4.38E-03	4.16E-04	3.68E-03	4.16E-04	1.75E-02	3.35E-03
Ovaries	1.3E-02[c]	2.5E-02[c]	1.92E-05	1.30E-05	2.68E-04	2.24E-04
Liver/gall bladder/ spleen	7.04E-03	5.86E-03	7.04E-03	5.86E-03	3.76E-02	2.90E-02
Urinary bladder	1.3E-02[c]	2.5E-02[c]	1.92E-05	1.30E-05	2.68E-04	2.24E-04
Colon/rectum	1.3E-02[c]	2.5E-02[c]	1.92E-05	1.30E-05	2.68E-04	2.24E-04
Testes	2.5E-03[c]	5.00E-03[c]	3.20E-06	1.30E-07	1.07E-05	4.30E-07
Lungs	7.04E-03	5.86E-03	7.04E-03	5.86E-03	3.76E-02	2.90E-02
Thymus	7.04E-03	5.86E-03	7.04E-03	5.86E-03	3.76E-02	2.90E-02
Esophagus	7.04E-03	5.86E-03	7.04E-03	5.86E-03	3.76E-02	2.90E-02
Stomach	7.04E-03	5.86E-03	7.04E-03	5.86E-03	3.76E-02	2.90E-02
Bone surfaces	7.04E-03	5.86E-03	7.04E-03	5.86E-03	3.76E-02	2.90E-02
Remainder	7.04E-03	5.86E-03	7.04E-03	5.86E-03	3.76E-02	2.90E-02
Female breast	8.16E-03	6.37E-04	8.16E-03	6.37E-04	3.67E-02	4.99E-03
Uterus	1.3E-02[c]	2.5E-02[c]	1.92E-05	1.69E-05	2.25E-04	2.24E-04
Bone marrow	1.18E-03	1.20E-03	1.18E-03	1.20E-03	8.13E-03	7.65E-03
Skin[d,e]	4.32E-02	1.76E-02	4.32E-02	1.76E-02	1.50E-01	6.02E-02

a. SID = 183 cm.
b. Image receptor size 35.6 cm by 43.2 cm.
c. Modified from Webster and Merrill (1957) as presented in ORAUT (2005a).
d. Calculated using backscatter factor of 1.35 from NCRP 102 (1989, Table B-3) for skin dose estimates through 1985, consistent with ORAUT (2005a).
e. Calculated using backscatter factor of 1.40 from NCRP 102 (1989, Table B-3) for skin dose estimates after 1985, consistent with ORAUT (2005a).

3.4 UNCERTAINTIES

As stated in ORAUT (2005a), *error* is defined as deviation from the correct, true, or conventionally accepted value of a quantity, and *uncertainty* is defined in terms of the potential range of a stated, measured, or assumed or otherwise determined value of a quantity. Error and uncertainty provide an indication of confidence in the dose estimates. Uncertainty, expressed in terms of a confidence level, is a more appropriate term than error, which implies that the actual value is known. Uncertainty, stated as a probability of falling within a stated range, includes precision and reproducibility of the measurement as well as accuracy (i.e., how close the estimate comes to the actual value).

Although many factors can introduce uncertainty and error into X-ray exposures, five factors contribute the most uncertainty to the dose estimate: (1) measurement error, (2) variation in applied kilovoltage, (3) variation in beam current, (4) variation in exposure time, and (5) SSD. Film speed, the use of screens, or the use of grids would not affect the beam output intensity. The lack of historical records for some of these measurements introduces uncertainty into the dose estimates that cannot be readily quantified, although there is no apparent reason to believe that practices at SNL/CA or its medical subcontractors were different from those at other facilities or from recommended standards of the medical community at the time. The following estimates of uncertainty associated

Table 3-5. Organ doses from lumbar spine X-ray, 1956 to 1971.

Organ	Estimated dose[a,b] HVL = 2.5 mm Al (collimated)	
	LAT (rem)	AP (rem)
Thyroid	3.88E-05	2.55E-04
Eye/brain	3.88E-05	2.55E-04
Ovaries	1.52E+00[c]	1.12E+00[c]
Liver/gall bladder/spleen	5.43E-02	6.72E-02
Urinary bladder	1.52E+00[c]	1.12E+00[c]
Colon/rectum	1.52E+00[c]	1.12E+00[c]
Testes	1.12E+01[c]	5.40E-02[c]
Lungs	5.43E-02	6.72E-02
Thymus	5.43E-02	6.72E-02
Esophagus	5.43E-02	6.72E-02
Stomach	5.43E-02	6.72E-02
Bone surfaces	5.43E-02	6.72E-02
Remainder	5.43E-02	6.72E-02
Female breast	5.04E-02	2.13E-02
Uterus	1.20E-01	2.44E-01
Bone marrow	8.54E-02	3.15E-02
Skin[d]	5.39E+00	1.18E+00

a. SSD = 63 cm for AP X-ray and 52 cm for LAT X-ray.
b. Image receptor size 35.6 cm by 43.2 cm.
c. Organ dose values for the testes and ovaries (and analogs) for lumbar spine reflect actual measurements reported in Lincoln and Gupton (1958).
d. Skin dose values include backscatter factors of 1.39 from Table B.8 of NCRP 102 (1989).

with X-ray exposures are from ORAUT (2005a), which this analysis relied on for default information when site-specific records of X-ray machine settings could not be found (specifically, for filtration and HVL values). Other values were taken from available notes in claim files and were selected to be favorable to claimants whenever a range of values was found in the claim files.

ORAUT (2005a) reports that X-ray doses are derived largely from actual measurements of X-ray machine output with R-meters or similar ionization chamber devices. Reportedly, these typically had an uncertainty of ±2% for photon energies below 400 keV if properly calibrated and used. Although more current machinery could have a smaller uncertainty, +2% is assumed to be conservative.

Variation in applied voltage generally falls within ±5% of the machine setting. Beam intensity is approximately proportional to the 1.7 power of the kilovoltage, resulting in an uncertainty of approximately +9% in relation to beam intensity for voltages in the 110- to 120-kVp range. Variations in tube current are normal and generally small. As the tube current drops, beam intensity falls in direct proportion. Large decreases in beam output would be readily detectable and would indicate the need for machine maintenance or, as a temporary measure, an increase in the current or voltage to provide the necessary intensity for proper radiography. ORAUT (2005a) estimates the variation in tube current to be approximately ± 5% for this parameter.

Exposure time can significantly affect the dose received from radiography (exposure times are a fraction of second). Even a small variation in exposure time due to timer error can significantly change beam output. Because early X-ray machine timers are known to have been inaccurate, ORAUT (2005a) assume uncertainty in beam output due to timers to be ±25%.

SSD can contribute to variability because the ESE is determined by this distance. Variations result from accuracy of positioning as well as worker size (thickness). As expressed in ORAUT (2005a), this is generally thought to vary by no more than a few centimeters, with an upper limit of 7.5 cm (±10%).

A potential source of uncertainty for SNL/CA is the number and type of X-rays taken. As noted above, reports indicate the performance of only an annual PA chest X-ray examination, but no official protocol has been found that would rule out the possibility of other X-ray views or more frequent chest examinations. At this time, dose reconstructors should assume a single annual PA chest X-ray for 1956 through 1989 [4]. Dose reconstructors should assign dose from the X-ray procedures actually listed on the pre-employment X-ray record form. If the claim file is missing the pre-employment X-ray record, the dose reconstructor should assign dose from a PA chest and AP and LAT lumbar spine as the default for the pre-employment X-rays.

Another source of uncertainty is the lack of site-specific values for filtration and exposure time. While the claim files show a consistent notation of kVp and SID, interpretation was required to determine the beam current exposure before 1985. For all time periods, Figure 3-1 was required to estimate air kerma. Filtration and HVL were not noted in claimant files so the estimated values used in this SPD were favorable to claimants and consistent with practices during each time period.

Consistent with ORAUT (2005a), this analysis relies on the statistical root mean square to estimate total uncertainty. The root mean square is the square root of the sum of the squares of the individual uncertainty values and equals 28.9% assuming all variation is in the positive direction. An estimate of 30% uncertainty is favorable to claimants.

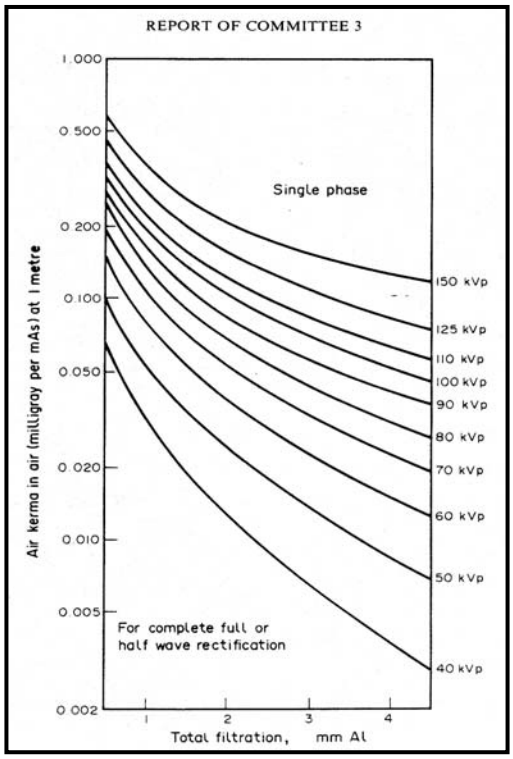

Figure 3-1. Kerma in air at 1 m from X-ray source as a function of total filtration for various values of tube potential (ICRP 1982).

4.0 OCCUPATIONAL ENVIRONMENTAL DOSE

4.1 PURPOSE AND SCOPE

This section addresses the occupational environmental dose applicable to the SNL/CA site from the beginning of operations (1956) to the present time. The term *occupational environmental dose* refers to the radiation dose received outside of buildings, but on the SNL/CA site, as a result of ambient airborne radionuclides or ionizing radiation.

4.2 INTRODUCTION

The SNL/CA site is located directly to the south of LLNL. The southern perimeter fence of LLNL is directly across the street from the northern perimeter fence of SNL/CA. Thus, environmental dose is addressed in terms of potential exposures that may occur on the SNL/CA site as a result of both SNL/CA and LLNL operations. Aside from operations at the TRL, the internal exposures on the SNL/CA are assumed to be due to LLNL operations alone. According to SNL/CA (1998), no other measurable effluents of radionuclides from SNL/CA have existed. The external dose rate at the LLNL south perimeter, measured by TLDs, has been reported from 1967 to the present. These measurements are also considered in estimating external dose for SNL/CA.

Internal and external exposures to radionuclides in the outdoor environment are considered separately in this section. Section 4.3 presents information necessary to estimate internal environmental dose; radionuclides of concern are first identified. The estimated source terms (release rates) for radionuclides considered potentially significant to internal environmental dose and internal exposure (yearly intake) are also addressed. Annual environmental reports for SNL/CA and LLNL form the basis of these estimates (LLL 1972 to 1982, 1984; LLNL 1986 to 2005; LRL 1961a,b, 1963 to 1971; SNL/CA 1986 to 1988, 1989b, 1990b, 1991d, 1992, 1993, 1994a, 1995 to 1998, 2005; SNLL 1982, 1983a, 1984).

Section 4.4 contains information necessary for estimating external environmental dose. Ambient external dose rates, reported in annual environmental reports for LLNL and SNL/CA, are used to estimate dose rates for the operational period of the SNL/CA.

Section 4.5 considers uncertainties in the information provided for estimating occupational environmental dose. The discussion addresses sources of uncertainty and provides quantitative information where possible.

4.3 INTERNAL DOSE FROM ONSITE ATMOSPHERIC RADIONUCLIDE CONCENTRATIONS

4.3.1 Radionuclides of Concern

The only detectable effluents from the SNL/CA site occurred during operations of the TRL, when ^{3}H was released (Sect. 2.4.2). Releases of DU from the SNL/CA site are not considered a significant source of environmental exposure at the SNL/CA. Such releases would only occur during machining operations, but according to SNL/CA (1990b), all operations at SNL/CA using DU were equipped with exhaust air systems with absolute filters at that time. Furthermore, machining operations, which began in 1971 (Adolphson 1972), were monitored closely for airborne contamination; operations were immediately stopped if airborne contamination was detected (SNL/CA 1989a).

However, the presence of measurable concentrations of tritium and airborne particulate radionuclides in the LLNL south perimeter area, nearly adjacent to the north perimeter area of SNL/CA, requires that these radionuclides be considered as potentially significant contributors to environmental dose on the SNL/CA site. In addition to ^{3}H, LLNL has processed and handled a number of radionuclides, including uranium and transuranic elements, mixed fission products, and accelerator-produced isotopes. Tritium, ^{239}Pu, and isotopes of uranium have been identified as radionuclides of significance for the ORAUT Environmental Occupational Dose Technical Basis Document (TBD) for LLNL (ORAUT 2005b), consistent with the list of radionuclides identified by the LLNL environmental monitoring program as representing more than 90% of the LLNL radioactive materials inventory.

Before considering any or all of the particulate radionuclides from the LLNL site as isotopes of significance at SNL/CA with respect to environmental dose, the concentrations at the LLNL perimeter nearest to SNL/CA and the potential associated doses were first evaluated. Between 1961 and 1970, concentrations of gross alpha and gross beta were measured at one or more perimeter locations near the south side of the LLNL site, as well at several offsite locations in the Livermore valley (LRL 1961a, 1963 to 1970). Isotope-specific data were not reported during this period. The data reported included contributions from naturally occurring alpha- and beta-emitting radionuclides. By comparing average off-site measured concentrations of gross alpha- and beta-emitting airborne particulate radionuclides with those measured at the southern perimeter, net concentrations of these particulates were calculated. The maximum net beta concentration during this period was 3×10^{-2} Bq/m^3; the maximum net alpha concentration was 5.9×10^{-5} Bq/m^3. When not zero (i.e., when the concentrations at the perimeter were not less than average measured off-site concentrations), committed organ doses associated with the alpha- and beta-emitting particulate radionuclides were calculated, assuming an annual inhalation rate of 2400 m^3, and that the alpha-emitting radionuclides were comprised wholly of ^{234}U, and the beta-emitting radionuclides were comprised wholly of ^{90}Sr (a fission product). Dose factors were taken from ICRP (2001) and were selected on the basis of which organ received the highest dose (i.e., the maximum organ dose was calculated). The representative radionuclides were selected based on their relatively higher organ dose factors compared to other possible representative radionuclides, such that the analysis is not likely to overlook potentially significant contributors to internal dose. A comparison of the gross-alpha and gross-beta dose calculations indicated that the organ dose associated with beta-emitting radionuclides generally contribute less than 5% of the committed dose, but always contribute less than 1 mrem to the yearly committed organ dose associated with environmental exposures for this time period. The committed doses associated with gross alpha for a 1-year intake ranged from 0 to 140 mrem, the latter based on the assumption that the “gross alpha” intake could be represented by ^{234}U.

From 1971 to the present, concentrations of particulate $^{239/240}Pu$, ^{235}U, ^{238}U, and gross beta-emitters have been measured at a south LLNL perimeter location directly across the street from SNL/CA (LLL 1972 to 1982, 1984; LLNL 1986 to 2004). Tritium was measured by LLNL at this location since 1973. At this location (labeled CAFÉ in LLNL annual environmental reports), the maximum net beta-emitter inhalation dose is estimated to be 0.03 mrem/yr, assuming that ^{90}Sr is the representative radionuclide. The maximum concentration of ^{239}Pu between 1971 and 2004 was 2.6×10^{-6} Bq/m^3, which corresponds to an inhalation intake of 6.3×10^{-3} Bq/yr, assuming a yearly inhalation rate of 2400 m^3/yr. The maximum committed organ dose from this annual intake is estimated as 1.1 mrem. At the same location, the maximum concentrations of ^{235}U and ^{238}U were 1.6×10^{-12} g/m^3 and 2.2×10^{-10} g/m^3, respectively. These concentrations correspond to annual intakes of 3.1×10^{-4} Bq/yr and 6.4×10^{-3} Bq/yr of ^{235}U and ^{238}U, respectively. Making the favorable to claimant assumption that these intakes are both of ^{234}U; the corresponding committed maximum organ dose for a one-year intake is 6.7 mrem.

These measured concentration data and dose calculations indicate that particulate airborne beta-emitting radionuclides originating from LLNL are not significant contributors to environmental occupational dose on the SNL/CA site. This is consistent with the finding in LLNL environmental reports, indicating that the gross beta activity is due to global fallout, and fluctuates in a manner typical of that source (LLL 1973 to 1982, 1984; LLNL 1986 to 2005). The alpha-emitting radionuclides ^{234}U and ^{239}Pu (represented as gross alpha for pre-1971), from LLNL, are retained as potentially significant contributors to internal environmental dose at SNL/CA. Inhalation intakes for these radionuclides are addressed in Section 4.3.3 below.

Tritium releases from the LLNL operations (Peterson 2005) and from the TRL are also considered as potentially significant contributors to intakes. Intakes are estimated according to the methods described in Section 4.3.3.

4.3.2 Source Terms for Internal Dose

No onsite environmental sources or outdoor concentrations of radionuclides have been reported for SNL/CA before the beginning of operation of the TRL in 1979. Release of radionuclides from activities at LLNL may have affected intakes by workers at SNL/CA, as noted in Section 4.3.1 above. The assumption that the nearest perimeter concentrations at the LLNL site represent the onsite concentrations at SNL/CA before operation of the TRL is favorable to the claimant, as a decrease in air concentration with distance from the LLNL perimeter would likely occur. From April through September the prevailing winds are from the west and southwest and variable throughout the remainder of the year (LLNL 1988); thus, this assumption is most favorable to the claimant during these months.

4.3.3 Annual Intake of Radioactivity

Inhalation intakes of airborne particulate ^{234}U and ^{239}Pu and of airborne ^{3}H are estimated in this section. Ingestion of drinking water contaminated by ^{3}H is also considered.

In order to calculate inhalation intake, it was necessary to consider the onsite air concentrations of ^{234}U, ^{239}Pu, and ^{3}H due to the presence of these isotopes near the LLNL south boundary (closest to the SNL/CA). Between 1979 and 1996, it was also necessary to consider on site ^{3}H concentrations resulting from TRL emissions.

Intakes were calculated by multiplying the relevant concentrations by an assumed inhalation rate of 2400 m^3/yr. For ^{3}H, the intakes also reflect skin absorption, as the inhalation intake was multiplied by a factor of 1.5. For airborne particulates (i.e., ^{234}U and ^{239}Pu), no particle size information was available, and thus the default ICRP 66 (ICRP 1994) value of 5 µm activity median aerodynamic diameter is recommended. Furthermore, no solubility information is available for airborne uranium or plutonium particulates, and thus assumed solubility should be selected based on what is most favorable to the claimant in light of the organ of interest. Tritium should be assumed to be associated with HTO, as this will give the highest dose associated with the two forms (T_2 and HTO) known to be released from both LLNL and SNL/CA (Garcia and Gorman 1996; Peterson 2005).

4.3.3.1 Inhalation Intakes

The following methodological information is summarized in Table 4-1. Calculated intakes are provided in Tables 4-2 and 4-3.

Table 4-1. Summary of methodology for estimating intakes of ^{3}H, ^{234}U, and ^{239}Pu).

Radionuclide	Applicable period	Method used
^{3}H	1956–1972	Multiply LLNL ^{3}H release (Peterson 2005) by average ratio of south perimeter ^{3}H concentration to release, derived from 1973-2003 data, to obtain estimate of maximum SNL/CA concentration (Bq/m^{3}). Multiply maximum concentration by 2400 m^{3}/yr inhalation intake rate, and by 1.5 to account for skin absorption.
	1973–1978 and 1997–2004	Reported south LLNL perimeter ^{3}H concentrations are used directly to estimate maximum SNL/CA concentration (Bq/m^{3}). Multiply maximum concentration by 2400 m^{3}/yr inhalation intake rate, and by 1.5 to account for skin absorption.
	1979–1996	Reported south LLNL perimeter ^{3}H concentrations are used directly to estimate LLNL contribution to maximum SNL/CA concentration (Bq/m^{3}). TRL contribution (onsite) derived by multiplying reported TRL releases by the maximum calculated ratio of near-TRL air concentrations to TRL releases for the years 1994 and 1995. Multiply both perimeter and near-TRL concentrations by 2400 m^{3}/yr inhalation intake rate, and by 1.5 to account for skin absorption. Add intakes at LLNL perimeter to intakes near TRL to account for both SNL/CA and LLNL contributions.
^{234}U	1956–1960	No date available for this time period.
	1961–1970	Reported south LLNL perimeter gross alpha concentration measurements were corrected for average Livermore valley background contributions, and multiplied by an inhalation intake rate of 2400 m^{3}/yr to estimate maximum SNL/CA intakes (Bq/yr).
	1971–2004	Reported south LLNL perimeter ^{235}U and ^{238}U activity concentrations (Bq/m^{3}) were multiplied by an inhalation intake rate of 2400 m^{3}/yr to estimate maximum SNL/CA intakes (Bq/yr). The unreported ^{234}U concentrations were addressed by assuming ^{234}U intake was equivalent to ^{238}U. Activity intakes for all radionuclides are added, and assumed to be ^{234}U.
^{239}Pu	1956–1960	No data available for this time period.
	1961–1970	Reported south LLNL perimeter gross alpha concentration measurements were corrected for average Livermore valley background contributions, and multiplied by an inhalation intake rate of 2400 m^{3}/yr to estimate maximum SNL/CA intakes (Bq/yr).
	1971–2004	Reported south LLNL perimeter ^{239}Pu activity concentrations (Bq/m^{3}) were multiplied by an inhalation intake rate of 2400 m^{3}/yr to estimate maximum SNL/CA intakes (Bq/yr).

^{3}H Intakes

Measured concentrations of ^{3}H at the south perimeter of the LLNL are available in LLNL annual reports from 1973 to 2004 (LLL 1973 to 1982, 1984; LLNL 1986 to 2004). These values are used to

estimate environmental intakes of ^{3}H for SNL/CA workers during these years without considering additional dilution in transport to the SNL/CA site, thus providing an estimate that is favorable to the claimant. Two LLNL annual environmental reports were not located (for 1982 and 1984); for these 2 yr, the ^{3}H concentrations were estimated to be the average of the preceding and following year's concentrations.

Table 4-2. Maximum site-wide annual median inhalation intakes (Bq/yr), 1956 to 1970.[a]

Year	^{3}H[b,c]	^{234}U or ^{239}Pu[d]
1956	1.2E+04	(e)
1957	4.0E+04	(e)
1958	2.1E+04	(e)
1959	1.2E+04	(e)
1960	1.1E+04	(e)
1961	3.9E+03	1.2E-01
1962	1.5E+04	1.4E-01
1963	2.1E+04	2.7E-02
1964	9.3E+04	6.2E-02
1965	1.3E+06	0.00[f]
1966	6.0E+04	0.00[f]
1967	2.3E+04	0.00[f]
1968	2.4E+04	0.00[f]
1969	2.6E+04	8.9E-03
1970	1.0E+06	0.00[f]

a. Assumes an inhalation rate of 2,400 m^3/yr.
b. Assumes total ^{3}H intake is the sum of inhalation and skin absorption, estimated by multiplying the inhalation intake by a factor of 1.5.
c. Intake of ^{3}H is based on historical LLNL source terms from Peterson (2005); see Section 4.3.3.1 and Table 4-1.
d. Gross alpha is reported for 1961 to 1970; recommend assuming the isotope of either ^{234}U or ^{239}Pu that gives highest dose to the organ of interest.
e. No measurements reported for these years.
f. A zero value indicates the measured concentration was less than or equal to the off-site background gross alpha concentration.

Before 1973, ^{3}H concentrations at the LLNL perimeter were not reported in annual reports. For these years (1956 to 1972), concentrations are estimated by multiplying the average ratio of perimeter air concentration of ^{3}H to reported annual LLNL ^{3}H released for the years 1973 through 2003 (found to be 2.7×10^{-14}) multiplied by the reported ^{3}H release for each year from 1956 to 1972. Tritium releases from LLNL are estimated back through 1953 by Peterson (2005). Tritium intakes from 1956 through 1970 are reported in Table 4-2. Intakes derived using this methodology for 1971 and 1972 are included in Table 4-3. After 1972, but for years when the TRL was not operating (1973 to 1978 and 1996 to 2004), the intakes were calculated based on the LLNL south perimeter concentrations and are given in Table 4-3.

Between 1979 and 1996, the environmental ^{3}H concentrations on the SNL/CA site were due to LLNL releases as well as the TRL releases. Measurements on the SNL/CA site near the TRL (where ^{3}H concentrations were expected to be highest) were not reported before 1994, due to the inability of SNL/CA monitors to detect environmental levels of ^{3}H at that time (SNL/CA 1994a). In 1994 and 1995, ^{3}H releases from the TRL were between 70 and 80% of the LLNL releases. Thus, although most of the measured ^{3}H near the TRL was likely to be due to TRL releases during these 2 yr, some of it would have been due to LLNL releases. Thus, the ratio of the measured ^{3}H concentration in air near the TRL to the ^{3}H release rate from the TRL in 1994 and 1995 is an overestimate of the true

ratio. Despite this, the maximum ratio derived for these 2 yr was $6.9 \times 10^{-13,}$ and this multiplier was used to estimate air concentrations near the TRL for the years where measurements were not available for this location (1979 to 1993).

Table 4-3. Maximum site-wide annual median inhalation intakes (Bq/yr), 1971 to 2004.[a]

Year	^{3}H[b,c]	^{234}U[c]	^{239}Pu[c]
1971	4.9E+03[d]	3.5E-03	6.3E-03
1972	4.8E+03[d]	6.0E-03	2.8E-03
1973	8.3E+03	5.5E-03	1.5E-03
1974	1.1E+04	4.8E-03	2.9E-03
1975	1.2E+04	4.6E-03	2.0E-03
1976	1.9E+04	5.4E-03	8.0E-04
1977	1.6E+04	7.3E-03	2.1E-03
1978	1.7E+04	9.8E-03	2.7E-03
1979	8.7E+03[e]	5.2E-03	1.5E-03
1980	9.6E+03[e]	3.6E-03	4.4E-04
1981	1.4E+04[e]	7.1E-03	1.4E-03
1982	2.6E+04[e]	8.0E-03	1.4E-03
1983	1.5E+04[e]	8.9E-03	2.1E-03
1984	2.0E+04[e]	7.1E-03	1.2E-03
1985	5.2E+04[e]	5.3E-03	1.8E-04
1986	7.2E+04[e]	4.7E-03	1.2E-04
1987	1.8E+05[e]	5.6E-03	5.3E-05
1988	1.5E+05[e]	1.3E-02	8.0E-05
1989	8.2E+04[e]	9.4E-03	1.0E-04
1990	2.9E+04[e]	5.7E-03	1.2E-04
1991	4.5E+04[e]	5.9E-03	1.1E-04
1992	2.7E+04[e]	5.1E-03	8.9E-05
1993	1.8E+04[e]	4.2E-03	8.6E-05
1994	9.0E+03	3.3E-03	8.2E-05
1995	2.8E+03	2.7E-03	5.9E-05
1996	5.8E+02	2.7E-03	5.8E-05
1997	4.7E+02	2.9E-03	1.6E-05
1998	3.3E+02	2.0E-03	1.4E-05
1999	2.3E+02	3.9E-03	1.3E-05
2000	1.5E+02	0.0E+00[f]	2.2E-05
2001	1.5E+02	0.0E+00[f]	9.2E-06
2002	1.7E+02	6.9E-04	7.6E-06
2003	2.0E+02	1.4E-03	6.0E-06
2004	9.2E+01	1.3E-03	7.4E-06

a. Assumes an inhalation rate of 2,400 m^3/yr.
b. Assumes total intake is the sum of inhalation and skin absorption, estimated by multiplying the inhalation intake by a factor of 1.5.
c. Intakes based on LLNL south perimeter concentrations; see Table 4-1.
d. Intake of ^{3}H is based on historical LLNL source terms from Peterson (2005); see Table 4-1.
e. Intakes include both LLNL south perimeter concentrations and TRL-generated concentrations; see Table 4-1.
f. Zero values indicate the analytical background exceeded the measured concentration of uranium isotopes.

In order to consider both LLNL sources and the TRL, the intakes estimated for the south LLNL perimeter were added to the intakes estimated near the TRL. Due to the varying relationship between the amount of LLNL ^{3}H releases and the TRL releases, it was not possible to assume

that the method of estimating concentrations near the TRL adequately considered LLNL contributions, especially for years when LLNL ^{3}H releases were 10 to 100 times greater than the TRL releases. Thus, intakes due to the LLNL south perimeter concentrations and the intakes due to the onsite concentrations near the TRL were added.

Particulate intakes (^{239}Pu and ^{234}U)
Measured concentrations of ^{239}Pu, ^{235}U, and ^{238}U at the south perimeter of the LLNL, near the SNL/CA site, were reported in annual environmental reports from 1971 through 2004 (LLL 1972 to 1982, 1984; LLNL 1986-2005). Intakes of ^{239}Pu for these years (Table 4-3) were calculated assuming the SNL/CA site concentration could be approximated by this south perimeter concentration. Intakes of uranium for these years (Table 4-3) were calculated by summing the intakes of ^{235}U and ^{238}U (Bq/yr) and an estimated ^{234}U intake associated with the perimeter concentrations. The ^{234}U intake was estimated by assuming it is equal to that of ^{238}U, which is approximately the case with natural uranium. The ^{235}U:^{238}U ratios on the LLNL main site perimeter have been reported as representative of natural uranium (LLNL 1994). The total activity intake of uranium is then assumed to be represented by ^{234}U, which is favorable to the claimant because most of the uranium activity (99%) is associated with ^{238}U and ^{234}U, and dose factors for ^{234}U are higher.

Before 1971, only gross alpha measurements were available (LRL 1961a, 1963 to 1971). From 1961 through 1970, the net alpha measurements, calculated by subtracting the average offsite gross alpha measurements (i.e., background values) from the reported values, were used to derive intakes of either ^{234}U or ^{239}Pu (Table 4-3). Net concentrations calculated to be less than zero were assumed to be zero. The assumed radionuclide should be the one which gives the highest dose to the organ of interest. For the five years of operation before 1961, there are no measurements available with which to estimate the intakes of ^{234}U and ^{239}Pu.

4.3.3.2 Ingestion Intakes

Ingestion of ^{3}H from drinking water available to workers was evaluated by considering the maximum site-wide intakes estimated for LLNL workers in ORAUT (2005b). It was noted in ORAUT (2005b) that several sources of drinking water for the LLNL site were sampled, along with water from the LLNL onsite swimming pool, which is close to the main sources of tritium at LLNL. The median activity in the drinking water sources was found to be below the limits of detection. However, concentrations in the pool were reported to range from 0.8 to 200 Bq/L; thus, the pool water was adopted as the site-wide maximum source of drinking water for LLNL workers (ORAUT 2005b). There is no reason to believe that SNL/CA workers would be exposed to drinking water at a higher concentration than the pool water at LLNL, so it was assumed here that the maximum concentration of ^{3}H in drinking water at any time might be represented by 200 Bq/L. This concentration would lead to an annual ingestion dose of less than 1 mrem/yr, regardless of whether the assumed form of ^{3}H is tritiated water or organically bound ^{3}H. This calculation assumes a 3 L/day ingestion rate and a 250-day work year (i.e., 750 L/yr ingestion of drinking water). Based on this calculation, ingestion of ^{3}H in drinking water is not a significant route of worker exposure.

4.4 EXTERNAL EXPOSURE TO ONSITE ATMOSPHERIC RADIONUCLIDE CONCENTRATIONS

4.4.1 Locations of Concern

Radionuclides present on the SNL/CA site with the potential to cause elevated ambient external exposures above background levels are restricted to those associated with the Radiography facility (SNL/CA 1993). An aerial survey of the LLNL and SNL/CA, conducted in 1975 (Tipton 1977), found

gamma-emitting radioactivity in excess of background levels in the vicinity of Building 9143 (the Radiography Building at the time) and to a much lesser extent, in the vicinity of a waste holding area known to contain DU and thorium (Building 9122), and a material storage vault known to contain DU, thorium, and small amounts of shielded ^{60}Co and ^{133}Ba, and ^{235}U (Building 921, incorrectly referred to as B-291 in the Tipton 1977 report). The highest estimated exposure rate at the center of the Radiography facility (as seen in the flyover) in excess of background gamma was 40 to 70 µR/hr, or 100-175 mrem/yr, for a 2,500 hr/yr occupational exposure. The highest values for the holding area and storage vaults were 2.0 to 8 µR/hr, or 5-20 mrem/yr for a 2,500 hr/yr occupational exposure. It was indicated in the 1977 annual monitoring report (LLL 1978) that none of the elevated areas posed a radiation hazard to workers. Furthermore, the survey indicated that the sources were restricted to work areas where access was limited. Thus, from the standpoint of environmental exposure, there were no areas identified in the survey containing radionuclides that posed a source of elevated ambient external exposure to workers.

As with the internal exposures, consideration must be given to external exposures arising from activities at the LLNL site in addition to those at the SNL/CA site. As early as 1964, LLNL measured perimeter external radiation (LRL 1965). Fluoroglass dosimeters were used at that time, with a limit of detection of 50 mrem. The reported dose rates were reported to be less than 0.01 mR/hr, which corresponds to less than 88 mrem/yr for continuous exposure. The use of TLDs apparently began in 1967 (LRL 1968). In 1971, a few perimeter locations were identified at which exposure rates were considered elevated above background (LLL 1972). One of these locations (Location 5) was adjacent to the LLNL cyclotron building and is at the south perimeter of the LLNL site and thus, directly across the street from the north perimeter of SNL/CA. Therefore, this LLNL perimeter location (south location) will be considered in evaluating external exposures. Environmental neutron measurements were also made at this location and reported in annual reports beginning in 1973. These are discussed in Section 4.4.3 below.

4.4.2 Gamma-Emitting Radionuclides

The use of TLDs to measure environmental radiation exposure at the SNL/CA site perimeter apparently began around 1989, as this was the first time that the five SNL/CA perimeter Thermoluminescent Dosimeters (TLDs) (Figure 4-1) are mentioned in annual environmental reports (SNL/CA 1990b). Before that time, the site relied on LLNL perimeter and off-site measurements, as the latter encompassed the SNL/CA perimeter (Figures 4-2 and 4-3). The estimated average and maximum dose external dose rates are listed in Table 4-4 and are based on both LLNL and SNL/CA perimeter measurements. The values in this table are applicable to an exposure duration of 2,500 hr/yr (50 hours per week, 50 weeks per year).

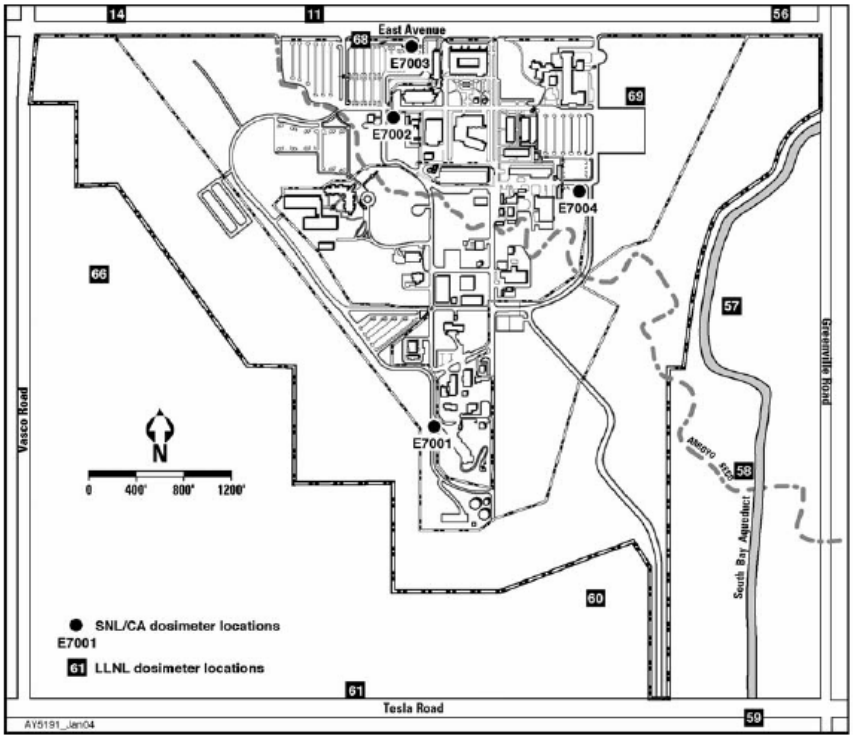

Figure 4-1. Thermoluminescent dosimeter locations since 1989.

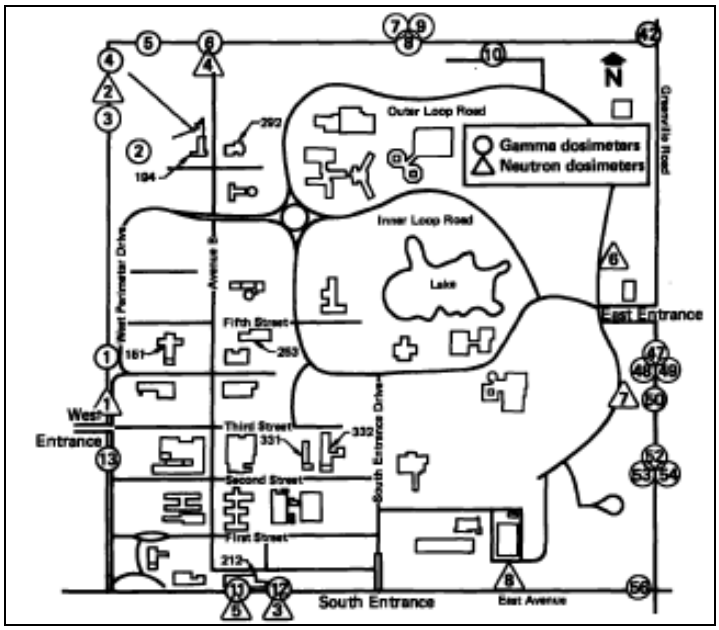

Figure 4-2. Location of LLNL perimeter gamma and neutron dosimeters (LLNL 1986).

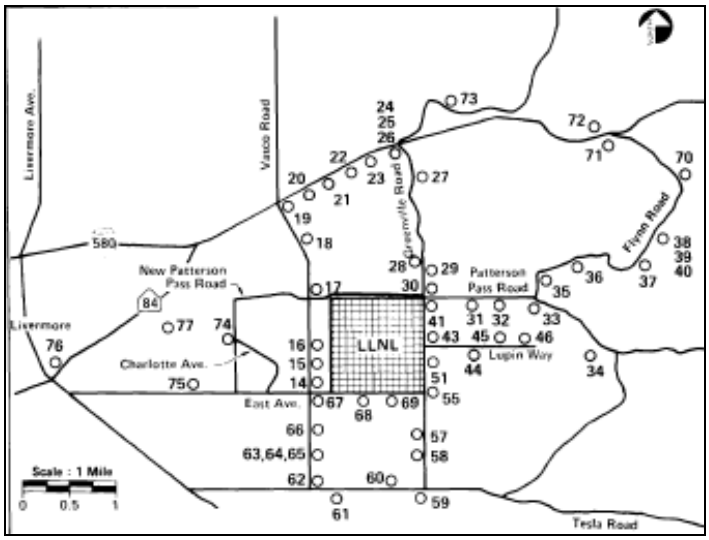

Figure 4-3. Location of LLNL offsite gamma dosimeters (LLNL 1988).

The data preference for this table was to include SNL/CA measurements when available, but this only occurred from 1990 to 1994 (SNL/CA 1991d, 1992, 1993, 1994a, 1995). Before 1990, south LLNL perimeter measurements were used to estimate SNL/CA exposure rates. This provides a maximum estimate of the contribution of LLNL exposure rates to the SNL/CA rates. This south perimeter value is also a reasonable estimate of the environmental exposure rates expected for the SNL/CA. The 1975 aerial survey indicated this measurement point coincided with the LLNL accelerator and thus, had a slightly elevated exposure rate on the order of that seen for the storage vaults and holding area at SNL/CA (Tipton 1977). Since most of the SNL/CA area surveyed in 1975 did not show exposure rates elevated above background, this is an assumption that is favorable to the claimant.

Table 4-4. External gamma radiation dose based on 2,500 hr/yr exposure duration (mrem/yr).

Year	Average perimeter[a]	Error[b]	Maximum dose rate[c]
1956	120[d]	36	156
1957	120[d]	36	156
1958	120[d]	36	156
1959	120[d]	36	156
1960	120[d]	36	156
1961	120[d]	36	156
1962	120[d]	36	156
1963	120[d]	36	156
1964	120[d]	36	156
1965	120[d]	36	156
1966	120[d]	36	156
1967	11	3	15
1968	120	36	156
1969	24	7	31
1970	40	12	52
1971	25	7	32
1972	35	10	45
1973	27	8	36
1974	39	12	51
1975	90	27	118
1976	84	25	109
1977	65	19	84
1978	27	8	35
1979	27	8	36
1980	23	7	30
1981	17	5	23
1982	19[e]	6	25
1983	21	6	27
1984	22[e]	7	28
1985	23	7	30
1986	17	5	23
1987	17	5	22
1988	16	5	21
1989	15	5	20
1990	15	4	19
1991	16	5	20
1992	16	5	20
1993	15	5	20
1994	16	5	21
1995	16[f]	5	20
1996	16[f]	5	20
1997	16[f]	5	20
1998	16[f]	5	20
1999	16[f]	5	20
2000	16[f]	5	20
2001	16[f]	5	20
2002	16[f]	5	20
2003	16[f]	5	20
2004	16[f]	5	20

a. Until 1990, unless otherwise noted, values are the south perimeter LLNL dose rate; after 1989, values are the average of SNL/CA perimeter dosimeters.
b. Error assumed to be ± 30% of the higher of the average perimeter (ORAUT 2006a).
c. Maximum dose rate is the average perimeter value plus error term.
d. Assumed maximum dose rate estimated for the average perimeter – no measurements reported for the LLNL south perimeter or for SNL/CA from 1956 through 1966.
e. Assumed the average of the year prior and year following – missing annual environmental reports for 1982 and 1984.
f. Assumed average of SNL/CA reported average perimeter values for 1990 to 1994.

There were no reliable measurements for either SNL/CA or the south perimeter LLNL location before 1967. The 1967 to 1969 values in Table 4-4 assume that the "experimental physics facility" mentioned in the corresponding annual reports (LRL 1968 to 1970) represents the south perimeter location that is labeled Location 5 from 1970 to 1980 (LRL 1971, LLL 1972 to 1981), but is relabeled Location 11 after 1980 (LLL 1982, 1984; LLNL 1986 to 2004). This is a reasonable assumption, since this location at that time corresponded to the location of the LLNL cyclotron.

The Table 4-4 exposure estimates for 1970 through 1980 correspond to measurements at LLNL's south perimeter TLD location 5 and, from 1981 through 1989, to TLD locations 11 and 12, with the exception of the 1982 and 1984 values, which were estimated by averaging 1981, 1983, and 1985 values due to the inability to locate annual reports for 1982 and 1984. Finally, SNL/CA TLD measurements are included for the years they are reported: 1990 through 1994. For these years, the "average perimeter" dose rate in Table 4-4 represents the average of the SNL/CA measurements reported. After 1994, the average perimeter dose rate is estimated by averaging the 5 yr worth of SNL/CA data.

Background radiation measurements for the Livermore valley have been reported in LLNL annual reports since 1971 (LLL 1972 to 1982, 1984; LLNL 1986 to 2004). Figure 4-4 is a summary of the quarterly results from 1988 through 1996, comparing LLNL average perimeter measurements and average LLNL Site 300 measurements to the off-site measurements. From Figure 4-4, it is evident that there have been only minor quarterly fluctuations around the value of 14 mrem/quarter, but no significant long-term trends of either the background measurements or the LLNL perimeter measurements. The average yearly background rate, calculated from the yearly rate for these 9 yr (LLNL 1997), is 57 mrem/yr for continuous exposure or 16 mrem/yr for a 2,500-hr/yr occupational exposure.

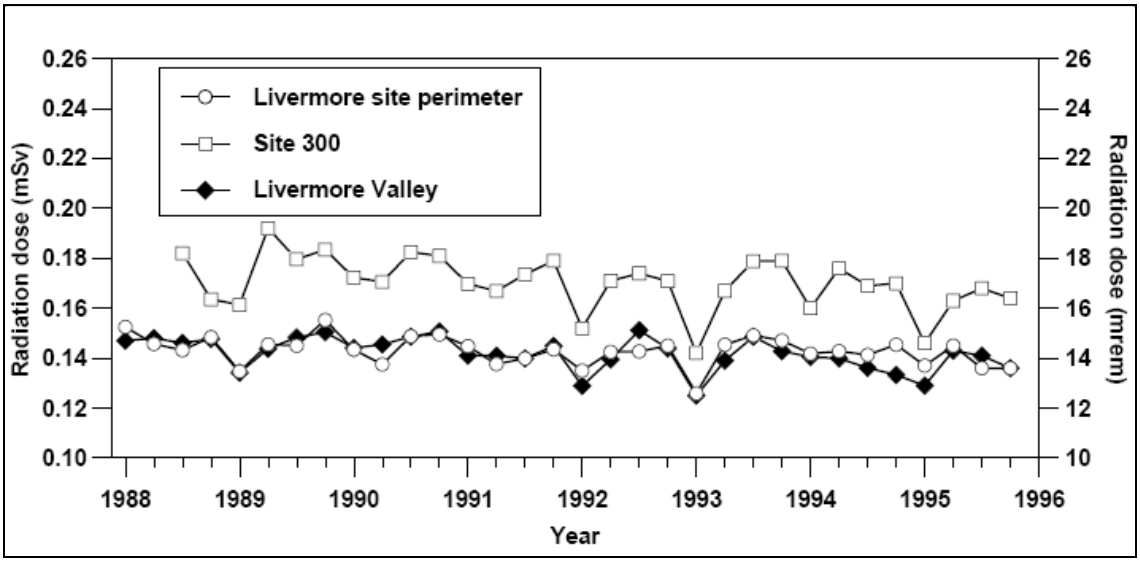

Figure 4-4. Livermore valley radiation background measurements in comparison with annual average measurements at LLNL site perimeter and LLNL Site 300 (LLNL 1997).

4.4.3 Neutron Exposure

The Insulating Core Transformer (ICT) (also known as Rotating Target Neutron Source I) accelerator began operation in 1966 in Building 212 at LLNL (Peterson 2005). This facility was known to be responsible for elevated levels of environmental neutrons at the LLNL south perimeter, across the street from the north perimeter of SNL/CA (LLL 1974 to 1982, 1984; LLNL 1986 to 1988). The increased neutron dose rate was attributed to operation of the ICT, also referred to as a 14-MeV neutron generator in LLNL annual reports. Environmental neutron monitoring began at LLNL in 1973, but was discontinued in 1994 since the responsible operations have been discontinued (LLNL 1996). Measurements were made at Location 5 (Figure 4-2) initially (LLL 1974), but a monitor at Location 3 (originally called 5a) was added in 1975 (LLL 1976). By 1987, the dose rates were at background neutron levels, which is approximately 4 mrem/yr for continuous exposure (LLNL 1995).

Although the dose rates were fairly elevated at the measurement location for several years, the dose rate at the SNL/CA perimeter would be considerably less due largely to dilution as the beam spread increases with distance from the source. The distance from the source to the monitoring locations on the LLNL perimeter was approximately 17 m (56 feet) (Willhoite 1979). Assuming that this is also the approximate distance to the nearest SNL/CA perimeter (which currently is across a 4-lane street, with shoulders on either side), the neutron dose rate at the SNL/CA perimeter would be approximately one-fourth the LLNL perimeter dose rate according to the inverse square law. The resulting estimated dose rates at the SNL/CA north perimeter are also corrected for an occupational exposure duration of 2,500 hr/yr and background contributions and are given in Table 4-5. These values are favorable to

the claimant, as the dose rate drops off fairly rapidly throughout other areas of the SNL/CA site. Due to the lack of information in years before 1973 regarding the operation of the ICT, it was assumed that the 1973 value was representative of all previous years of operation (i.e., 1966 to 1972).

Table 4-5 Elevated neutron dose due to LLNL perimeter neutron source, based on 2,500 hr/yr exposure duration (mrem/yr)[a].

Year	North perimeter dose[b]	Error[c]	Maximum dose[d]
1966	18[e]	5	23
1967	18[e]	5	23
1968	18[e]	5	23
1969	18[e]	5	23
1970	18[e]	5	23
1971	18[e]	5	23
1972	18[e]	5	23
1973	18	5	23
1974	26	8	34
1975	50	15	65
1976	43	13	55
1977	39	12	51
1978	9	3	12
1979	13	4	16
1980	6	2	8
1981	2	1	3
1982	8[f]	2	10
1983	8	2	10
1984	8[f]	2	10
1985	<1	<0.3	<1
1986	2	1	3
1987	<1	<0.3	<1

a. Reported for years neutron dose rate was believed, or unknown, to have exceeded background levels.
b. Value for south LLNL perimeter dose rate at Location 3 or 5, corrected for distance to SNL/CA, background, and assumed exposure time of 2,500 hr/yr.
c. Error assumed to be ± 30% of the average perimeter dose (ORAUT 2006a).
d. Maximum dose rate is the average perimeter value plus error term.
e. Assumed value reported for 1973 applied to previous years back to 1966.
f. Assumed the 1983 value – missing annual environmental reports for 1982 and 1984.

5.0 INTERNAL DOSIMETRY

5.1 SUMMARY OF POTENTIAL INTERNAL RADIONUCLIDE EXPOSURES

Work with radionuclides that created a potential for internal exposure at SNL/CA included limited uranium operations primarily (but not exclusively) DU machining, and research conducted in the Tritium Research Laboratory (TRL). Radiation workers were monitored using external dosimetry and urine bioassay. A "Radiation Worker" was defined as an "employee who receives or could potentially receive radiation exposure from his job in excess of 10% of the applicable standards for internal or external exposure" (Wright 1979a). For the purpose of internal dose, individuals working with radioactive materials that present a potential internal exposure were included as radiation workers (Wright 1979a). The numbers of radiation workers based on internal exposure varied over the years. Twenty-one individuals were defined as radiation workers for 1980 based on potential tritium exposure.

5.1.1 Tritium Research Laboratory

The TRL was established in 1976 to perform research and development functions for the DOE Office of Defense Programs to support weapons development. The TRL at its peak operation employed approximately 35 experimenters and staff support personnel (Garcia and Gorman 1996).

The building was divided into two zones, (1) an office area and (2) a radioactive materials area. The zones were separated by two sets of double doors. The room air in the radioactive materials area was continuously monitored for tritium. The monitoring systems were set to alarm at specific action levels but there is no indication that the tritium monitoring data were ever used to estimate worker intakes. The TRL operations were generally concerned with the physical and chemical characterization of tritium and its compounds. Fabrication of tritium compounds for use as engineering components was also part of the mission of the TRL. All operations involving gram quantities of tritium were conducted inside gloveboxes. The building had a control room where data from the various monitoring devices were stored. This provided for real time monitoring of facility conditions (Wright 1981a).

Tritium in quantities greater than 0.1 grams was doubly contained in gloveboxes, special Sandia-designed containers, or Department of Transportation-approved containers. The Safety Analysis Report (Wright 1981a) anticipated that 50 grams of tritium might be handled in one doubly contained system and that the total tritium inventory in the facility would be approximately 300 grams.

Tritium handling operations were terminated in 1992. Cleanup activities were conducted at TRL from January 1992 to December 1995. After 1996, the TRL complex was converted to the CRDL (Garcia and Gorman 1996).

5.1.2 Uranium Alloy Machining

Uranium alloy machining was performed at the SNL/CA facility starting before 1972. Several memos indicate that machining and testing of uranium alloys had previously been performed at the Union Carbide Y-12 plant and elsewhere. However, requirements for SNL/CA were such that they needed "between 50 and 100 specimens of various shapes and sizes per month," (Adolphson 1972), thus requiring initiation of an on-site program.

The radiation safety requirements for the machining operation are described in SOP *Machining Depleted Uranium Metal* (SNL/CA 1989a). The depleted uranium is designated in the SOP as D-38.

The radiation safety requirements include air sampling and urine bioassay (SNL/CA 1989a). Daily air samples were required during all machining operations even though air-sampling data had shown that neither wet machining nor "burning (D-38) chips" (SNL/CA 1989a) resulted in airborne particulates. Urine bioassay was required semi-annually.

In addition to inhalation exposures, machining of D-38 posed a potential hazard from cuts and splinters. All such incidents were to be reported to the Medical Department and wound counts made on any puncture wounds. No data on wound counts were found in the records available to the authors.

The Uranium Tritide Bed installed at the TRL in 1991 also presented a potential for exposure to uranium powder [TRL, SOP No. 757 (SNL/CA 1991a)]. The operating procedure did not specify bioassay requirements. In addition, parts contaminated with UO_2 dust were received by SNL/CA (Lovell 1982) posing a potential airborne dust hazard.

5.2 BIOASSAY PROGRAMS

The only radionuclides of concern for internal dosimetry at SNL/CA appear to have been tritium and uranium. Tritium exposure occurred primarily in the TRL. However, tritium is ubiquitous in the natural environment as it is a cosmogenic naturally occurring radionuclide and was produced by atomic weapons testing (Turner, 1995). Tritium was also present in the environment due to activities at the LLNL facility adjacent to SNL/CA.

The primary detection method for intakes of ^3H at all SNL facilities has been urine bioassay (Potter ca. 1997). There is no evidence that urine bioassay samples were analyzed for any radionuclides except H-3 and natural uranium (U-nat), or that other types of bioassay, i.e., fecal analyses or in vivo counting, were employed at SNL/CA. The H-3 bioassay was limited to TRL experimenters and staff. The uranium bioassay was performed on individuals involved in machining DU as well as others involved with handling uranium powders or those in areas where air concentrations potentially exceeded 10% of the air concentration guidelines (Wright 1979a). Records indicate that tritium bioassay was performed at SNL/CA only after the TRL became operational.

5.2.1 Tritium Bioassay Programs

According to (Garcia and Gorman 1996), tritium bioassays were performed weekly for individuals involved in experimental work at the TRL from 1979 through 1995 when the TRL was decommissioned. Individuals working in the Waste Handling Facility may also have participated in the tritium bioassay program, at least during 1991 (Garcia 1991a, 1991b). Bioassays were also required for all personnel inside the TRL when an evacuation alarm occurred (SNL/CA 1991b). Additional samples were required in some cases by Safe Work Permits (SWPs) or at the discretion of Health Physics.

The *Tritium Research Laboratory Safety Analysis Report* (Wright 1981a) specifies weekly bioassay with samples analyzed by liquid scintillation counting. Laboratory analyses were performed, in house, by the Health Physics Division. Doses were calculated and reported on a monthly basis.

A 1993 memo from Donn Wright to Lydia Perez (Wright 1993) describes the methods by which tritium bioassay data were managed between 1977 and 1993. According to Wright, bioassay results were originally maintained in a VAX text file. The VAX files were transferred into REFLEX, a database management program in 1988. In 1990, the files were sent to SNL/NM for entry into personnel

dosimetry histories. As of 1993, the date of the memo, the bioassay data were collected by SNL/CA and a hard copy sent to SNL/NM monthly or quarterly.

The TRL Health Physics Quarterly Summaries from 1988 through 1995 confirm that urine specimens were collected on a weekly basis and tritium concentrations determined by liquid scintillation counting. Initially absorbed doses were calculated for all individuals whose urine bioassay results indicated that they might have received a dose greater than 10 mrem per calendar quarter based on the fact that 10 mrem was the reported sensitivity of the TLDs used for external dosimetry. From 1991 through 1992, absorbed doses were calculated for individuals whose bioassay results indicated a dose greater than 2 mrem per calendar quarter (Garcia 1991b). From 1993 through 1995, absorbed doses were calculated for individuals whose bioassay results indicated a dose greater than 1 mrem per calendar month (Garcia 1993a).

5.2.2 Uranium Bioassay Programs

Uranium bioassays were required for SNL/CA workers involved in DU machining and other operations where airborne uranium might have been encountered. The urine bioassay criteria for DU were described in Wright (1979a). The criteria for minimum routine (semi-annual) uranium bioassay were as follows:

- When air sampling results show concentrations at or greater than 10% of the concentration guide of 1E-10 $\mu Ci/cm^3$.
- For routine handling of uranium hydrides, solutions of uranium compounds, and uranium powders (>4 times per quarter).
- Machining of uranium.
- During any operation that the Hazards Control Division deemed hazardous, or requires air sampling in a SOP or SWP.

Nonroutine bioassays were performed in the following situations:

- Cut or lesion during handling or machining of uranium.
- An individual in close proximity or exposed to a uranium metal fire.
- Skin contact with a solution of uranium.

DU alloy machining was performed under SOP 1066 (SNL/CA 1989a). The SOP required that all machining operations be performed wet, reducing the risk of fire and generation of airborne dust. Revision C, dated October 1, 1989 (SNL/CA 1989a), required semi-annual urine bioassay for uranium but noted that “more frequent urine samples are usually collected.”

Air sampling was also required during machining. The machine operators were charged with the responsibility for turning on and off the air samplers. No air concentration data were available to the authors.

5.3 BIOASSAY LABORATORY PROCEDURES AND MINIMUM DETECTABLE INTAKES

The bioassay measurements were performed by various different entities during the period of operation of SNL/CA. No specific laboratory procedure manuals were available for either tritium or uranium analyses at SNL/CA.

5.3.1 Tritium

Tritium bioassay was performed at SNL/CA during the entire history of the TRL by liquid scintillation counting. Guidelines for the bioassay program are contained in Health Physics Operating Procedure 471583. Note: That document was not available to the authors as of the time this SPD was submitted.

Revision 3 of the TBD for internal dosimetry at SNL/CA and SNL/NM states that workers were instructed to fill the entire 1500 mL urinalysis container (Potter ca. 1997). However, a single void urine sample was prescribed for tritium. It should be noted that Revision 3 of the TBD was produced after the TRL operations ceased at SNL/CA. (The earliest date for the document would have been 1997 based on the dates of the references listed.)

The frequency and minimum detectable activities (MDAs) for tritium bioassay, gleaned from the available quarterly Health Physics Reports and Standard Operating Procedures are given in Table 5-1.

Table 5-1. MDAs for and frequency of tritium bioassay.

Year	MDA	Frequency	Source
1979–1983	Approximately 10 nCi/L (background +/-2 std. dev.)	Weekly	TRL Health Physics Summary, 1979-1983 (Lovell, Wright, and Hafner 1984)
1980	10 nCi/L,	Weekly	TRL Health Physics Summary, First Quarter 1980 (Hafner 1980)
1980	20 nCi/L		Calculated from liquid scintillation counter efficiency and background information in ^{3}H dose worksheets for 1980
1981	Approximately 10 nCi/L (background +/- 2 std. dev.)	Weekly	TRL Health Physics Summary, First Quarter, 1981 (Wright 1981a)
1986	Approximately 20 nCi/L		Calculated based on computer printout for positive exposures for 1986
1988	None given	Weekly	Tritium Research Laboratory Health Physics Quarterly Summaries (October) (HCD 1988)
1989	None given		Tritium Research Laboratory Health Physics Quarterly Summaries (HCD 1989a,b,c,d)
1990	None given	Weekly	Tritium Research Laboratory Health Physics Quarterly Summaries (Garcia 1990a,b,c,d)
1991	None given	Weekly	Tritium Research Laboratory Health Physics Quarterly Summaries (Garcia 1991c,d,e,f)
1991	None given. Reporting limit of 1 µCi/L specified.	Minimum frequency – monthly: weekly for individuals working primarily in the TRL	SOP No. 709, Tritium Research Laboratory Building 968
1992	None given	Weekly	Tritium Research Laboratory Health Physics Quarterly Summaries (Garcia 1992a,b,c,d)
1993	None given	Weekly	Tritium Research Laboratory Health Physics Quarterly Summaries (Garcia 1993a,b,c,d)
1994	None given	Weekly	Tritium Research Laboratory Health Physics Quarterly

			Summaries (Garcia 1994a,b,c,d)
1995	None given	Weekly	Tritium Research Laboratory Health Physics Quarterly Summary for First Quarter (Garcia 1995)
1998	5.5 nCi/L		Wright 1998 Memo to Debbie Miller DOE/AL Special bioassay

According to Hafner (2006), tritium bioassay samples were counted in a liquid scintillation counter under a standard protocol that required a 10-minute count. The volume of urine to be used in any sample was not specified but is assumed to be a minimum of 0.5 mL based on information in available dose calculation forms. The background count in the tritium channel was reported by Hafner to be approximately 22 cpm. Calibration forms for 1995 give background counts in the tritium channel as approximately 15 cpm with counting efficiencies of approximately 0.67. Daily performance checks from 1993 to 1995 showed efficiencies in the range of 0.65 and background counts from 17 to 21 cpm. Based on these parameter values, the minimum detectable concentration (MDC) for tritium in urine would be approximately 13 nCi/L.

$$MDA = (4.66Sb + 2.71)/(E*A*R*V) \quad \text{[Vosburg 1993]}$$

where:

Sb = blank standard deviation
E = efficiency
A = decays per disintegration = 1 for tritium
R = chemical recovery (not applicable for liquid scintillation counting for tritium)
V = volume of the sample = 0.5 mL

MDA = 13.5 dpm

Assuming a 0.5 mL aliquot, the MDC would be 29,000 dpm/L or 13 nCi/L. This is consistent with the MDA of 10 nCi/L reported in the 1980 and 1981 TRL Health Physics Summaries (Hafner 1980; Wright and Hafner 1981). However, a printout of positive exposures for 1986 shows efficiencies between 0.3 and 0.4 and background approximately 20 cpm. While the volume of the aliquot used is not specified, a review of the data in the printout supports the assumption that aliquots of 0.5 mL were used in the analyses. This would lead to an MDC of approximately 20 nCi/L.

Tritium monitoring results were generally reported in dose rather than urine concentration or estimated intake. However, bioassay appointment cards used during the period April 1982 through October 1982 provided the results of the bioassay and the calculated dose.

The SNL/CA personnel records reviewed generally report calculated dose for tritium rather than urine bioassay data. Chronic Dose Worksheets for 1980 indicate that tritium doses for SNL/CA workers were calculated based on bioassay results as follows:

$$Dose = (0.286)(\Delta t)(\mu Ci/L)$$

The origin of the constant, 0.286, is not defined, and Δt is assumed to be the period of time for which dose is calculated. The dose formula is included here to allow back-calculation of urinary tritium concentrations and tritium intake from dose data.

5.3.2 Uranium

Uranium bioassay samples were sent to SNL/NM for analysis either in house or by a contract laboratory (SNL/CA 1965-1990, Potter 1994). In general, bioassay procedures at SNL/NM required 24-hour urine sample collection. However, urine bioassay questionnaires that were completed by monitored workers at SNL/CA in 1989 and 1990 noted that approximately 50 mL were needed for analysis (SNL/CA 1965-1990). The form contained very brief instructions for sample collection. In contrast, the SNL/NM bioassay kit for 1993 included detailed instructions on how to obtain and deliver the sample (SNL/NM 1993a). Presumably since the uranium bioassays were processed by SNL/NM, the SNL/CA kits would have been the same for that period.

The analyses of uranium bioassays were performed by various different laboratories over the course of uranium processes at SNL/CA. For most of the time, the SNL/NM Industrial Hygiene Laboratory performed the uranium bioassay by either fluorimetry or by inductively coupled plasma mass spectrometry (ICP-MS). For several years during the late 1980s, Thermo-Analytical Incorporated/EAL Corporation (TMA/EAL) performed the uranium bioassay. Controls for Environmental Pollution (CEP) performed analyses during the early 1990s (SNL/NM 1993a, 1993b). However, subsequent inter-comparison studies indicated that the data from CEP were not reliable (Ball et al. 1995). Therefore, the DR should not use urine bioassay data from CEP analyses in the dose reconstruction. The timeline for uranium bioassays of SNL/CA workers is given in Table 5-2. The information in the table was gleaned from individual employee bioassay reports and Wright (1979a).

The results of urine bioassay were generally reported in mass concentration units. There is no information to demonstrate that appreciable amounts of either natural or enriched uranium were used at SNL/CA. Therefore, the DR should assume a specific activity for DU when converting mass concentration to activity concentration. The most reasonable specific activity for depleted uranium is 4.38E-7 Ci/g as stated by Wright (1979a).

Wright (1979a) states that the minimum detectable limit by the fluorimetric analysis method used by SNL/NM was 1E-2 µg/L or 4E-3 µCi/L. These two values are inconsistent, assuming the specific activity of 4.38E-7 Ci/g for depleted uranium also stated by Wright. A mass concentration of 1E-2 µg/L would be equivalent to an activity concentration of 4.38E-9 µCi/L. A 1975 analytical report from the Livermore Medical Department shows a detection limit for uranium of 0.005 µg in a 50 mL sample by an unspecified method, indicating a MDC of 0.1 µg/L (SNL/CA 1975-1977). Other sample data sheets show minimum detectable amounts of 0.01 µg (SNL/CA 1975-1977). With an aliquot presumed to be approximately 15 mL (the record was nearly unreadable), this would indicate an MDC of approximately 0.7 µg/L.

Table 5-2. Preliminary time line uranium in urine analyses.

Period	Laboratory	Method	MDA	Basis and comments
<1968	No information	No information	No information	No information
1968–unknown	Radiation Detection Company (RDC)	Fluorimetry	5 µg/L	Reported error (+/- 2σ)
1973–1986	SNL/NM Industrial Hygiene Services	Fluorimetry Methods SAND 75-0014 also reported on the data sheets.	Based on lowest reported levels, and the prior MDA , the MDA applicable to this period would most likely be in the range of 5 µg/L	Results reported as 0 in most cases with the lowest reported non-zero values of 1.1 µg/L to 3.2 µg/L appearing occasionally in the records. Retests were requested for positive results
1986	SNL/NM Industrial	ICP listed as	No information;	It is unlikely that the method was

	Hygiene Services	Procedure No.	all results reported as 0	inductively coupled plasma mass spectrometry. It did not come into general use until the early 90s. ICP alone is not sensitive.
1986 to 1987	TMA/EAL	Not specified but probably fluorimetry	"<" values ranged from 3 to 5 µg/L	"<" values provided in the reports
1987–1990	SNL/NM Industrial Hygiene Services	Fluorimetry Method SAND88-1149 also referenced	"<" values ranged from 10 to 12 µg/L	Notes on the reports
1990–1993	CEP		No MDC given	Audit memos show that CEP analyses were problematic.
1993-present	SNL/NM Industrial Hygiene Services	Not defined but probably ICP-MS	0.1 µg/L	Reasonable MDAs for ICP-MS

5.4 BIOASSAY RESULTS

Garcia and Gorman (1996) provide data on doses calculated from urine bioassay of TRL experimenters and staff. However, he provides no information on how the urine tritium concentrations or intakes were estimated. For the early years of operation (1979 to 1981) the only data available are the number of workers with doses within specified ranges as shown in Table 5-3.

Table 5-3. Range of tritium doses based on bioassay, 1979 to 1981 (Garcia and Gorman 1996).

Year	Number of workers receiving:	
	10–100 mrem/yr	**101–500 mrem/yr**
1979	4	1
1980	4	1
1981	6	0

For the years from 1982 through 1995, (Garcia and Gorman 1996) shows maximum and average doses from intake of H-3 at TRL as well as total person-mrem (Table 5-4). The number of workers monitored can be inferred by dividing the total person-mrem by the average dose in mrem.

Table 5-4 Estimated tritium doses based on bioassay, 1982 to 1995 (Garcia and Gorman 1996).

Year	Maximum Dose (mrem/y)	Average Dose (mrem/y)	Total Person-Dose (mrem/y)	No. of monitored Individuals (inferred)
1982	70	30	183	6
1983	79	49	148	3
1984	1620	152	3040	20
1984*	234	75	1420	19
1985	347	65	2270	35
1986	229	67	1330	32
1987	178	42	580	14
1988	218	63	1652	26
1989	232	46	2465	54
1990	262	30	1056	35
1991	111	11	458	42
1992	53	11	222	20
1993	63	17	257	15
1994	69	15	190	13
1995	42	15	134	9

*Maximum, average, and total person-dose excluding maximum individual single event

5.5 METHOD OF CALCULATING DOSE FROM URINE BIOASSAY DATA

In most cases, the information available in the records generally provides only the end result of the calculation, not the actual urine bioassay concentration data. The methods of calculating doses would have varied over time as more metabolic information and data on retention and urinary excretions of the radionuclides of interest became available.

The SNL/CA personnel records reviewed generally report-calculated dose for tritium rather than urine bioassay data. Chronic Dose Worksheets for 1980 indicate that tritium doses for SNL/CA workers were calculated based on bioassay results as follows:

$$\text{Dose} = (0.286)\ (\Delta t)\ (\mu\text{Ci/L})$$

Again, the origin of the constant, 0.286 is not defined.

5.6 AIRBORNE RADIONUCLIDE CONCENTRATIONS

Work in the TRL was conducted in sealed gloveboxes and/or in gloveboxes used in high-velocity air hood (HVAH) mode. The TRL used two systems for decontaminating glovebox air before release to the environment. The GPS removed tritium from sealed gloveboxes. The Vacuum Effluent Recovery System (VERS). The TRL high flow ventilation provided 6 to 10 room air changes per hour (SNL/CA 1991b).

Tritium monitors were installed in TRL. The monitors were used to detect the release of tritium into room air, gloveboxes, the stack, and to monitor the performance of the GPS/VERS (SNL/CA 1991b). The monitors activated audible/visible alarms at the monitoring point and in the TRL control room. At least one operating tritium monitor was required in each room.

According to the Quarterly Reports, the tritium monitoring system was continually upgraded during the 1988 through 1995 period. A planned new tritium monitoring system was cancelled in 1992 (Garcia 1992a). Monitors were installed in the hallways of the TRL. Double monitoring was conducted in each laboratory. As of 1988, sixty tritium monitors were in use (Hazards Control Division 1989a, 1989b, 1989c, 1989d).

The tritium monitors had Room Air Low and Room Air High Alarms. The Room Air Low alarm was triggered at a concentration of 30 µCi/m3. The High Alarm was set at 1 mCi/m3. The maximum permissible concentration in air (MPC_a) and the more recent derived air concentration (DAC) remained constant at 2E-5 µCi/mL (20 µCi/m3) throughout the operational history of the TRL. The alarms were checked on a routine basis. If a Room Air Low alarm was triggered, workers were required to mitigate the release within a few minutes and exit the room if the air concentrations did not decrease. A Room Air High alarm required evacuation from the room within one minute (SNL/CA 1991b).

Airborne uranium dust concentrations were measured in the areas where DU was machined. No data on measured concentrations or action levels were available to the authors.

5.7 UNMONITORED DOSE ESTIMATES

The vast majority of SNL/CA workers were employed in areas with little or no potential for intake of radionuclides generated at their work areas (SNL/CA 1979, Bryson 1972). Therefore, only a small

proportion of workers were required to participate in the bioassay program. Individuals who worked in the laboratory areas as experimenters or support personnel may have been exposed to airborne tritium (SNL/CA 1991a, 1991b). Machinists and others who worked in areas where DU was machined or other uranium operations were conducted may have been exposed to airborne uranium dust (SNL/CA 1989a).

If no bioassay records are available for claimants who are on record as having worked in the TRL or uranium machining areas, potential unmonitored doses or intakes can be calculated based on 10 percent of the airborne radionuclide concentrations limits since procedures required monitoring for individuals who could have potentially received internal doses in excess of 10 percent of the dose limit (Wright 1993) or when air sampling results demonstrated that air concentrations could reach or exceed 10 percent of the concentration guides for air (Wright 1979b). The applicable MPC_as in air, by year, are given in Table 5-5.

Table 5-5. MPC_as for tritium and uranium.

Date	MPC_a or DAC (40-hr week)		Source
	^{3}H (water)	^{238}U	
1959	2 E-5 µCi/mL	1E-10	NBS Handbook 69 (NBS 1959)
1990s	2 E-5 µCi/mL	2E-11	10 CFR 835, Appendix A

The calculated annual intake for a worker inhaling tritium at an MPC_a of 2E-5 µCi/ml would be 5E4 µCi. The calculated annual intake for a worker inhaling uranium at the MPC_a of 1E-10 µCi/ml would be 0.24 µCi; at 2E-11µCi/ml, the annual intake would be 0.05 µCi assuming a breathing rate of 10 cubic meters per day for 240 days per year (2.4×10^9 ml/year). A concentration of uranium in air of 10×10^{-10} µCi/ml multiplied by the breathing rate would result in a total annual intake of 0.24 µCi.

Because SNL/CA is located in close proximity to LLNL, it is not the only source of airborne radionuclides. Intakes for workers outside those facilities can, if necessary, be calculated based on environmental air concentrations.

5.8 RADIATION DOSIMETRY REPORTS

The AEC required annual dose report summaries for workers (Burke 1969). Records of annual dose reports for 1958 through 1967 were reviewed. The reports from 1964 through 1968 and 1973 showed no internal body depositions as determined by bioassays or exposure to airborne materials that resulted in internal body deposition. The reports for 1962 and 1963 specified no internal depositions in excess of one-half the body burden. Between 1958 and 1962, the reports state that there were no exposure resulting in internal body deposition of radioactive materials. The reporting form changed in 1976; subsequent reports for 77 and 78 did not show internal exposures.

The Personnel Monitoring and Laboratory Services department at SNL/NM issued monthly ALARA Radiation Dosimetry Reports that list all employees whose annual dose equivalent exceeds specific action levels of 2%, 6%, and 10% of the applicable 10 C.F.R. pt. 835 limiting values (Personal Monitoring and Laboratory Services 1997). The reports include external doses as measured by TLD badges and internal doses as committed effective dose equivalent (CEDE). The column for CEDE remained blank in most cases, presumably where no bioassays were performed. It is not clear whether these reports included SNL/CA employees.

Pre-1973 external doses were documented by SNL/CA. Tritium doses acquired at sites other than SNL were noted occasionally in the records (SNL/CA undated-c). The notation showed values less

than 3 µCi/L presumably in urine. One record in this undated document showed 3 MC/L; however, the entries in this record were all in capital letters, and most likely were most likely meant to be 3 µCi/L.

Termination Occupational Exposure Reports were also filed for SNL/CA employees. These reports were primarily focused on external exposures but did include a section for internal exposure (SNL/CA 1980-1984).

5.9 SUMMARY TABLES

The following summary tables for MDAs incorporate the best general information available. However, it should be noted that some individual data in the records reviewed showed lower MDAs or reported urine bioassay values less than the MDA listed.

Table 5-6. Internal dose control program.

Routine Monitoring Type	Period	Frequency
Urine – tritium	1977–1995	Weekly
Urine – uranium	1968–present	Semi-annual or as required by the Hazards Control Department

Table 5-7. Detection limits for urine bioassay for ^{3}H with liquid scintillation counting.

Period	MDA (pCi/L)
1977–1983	10,000 (reported)
1980 - 1985	20,000 (calculated for 1980 based on background and efficiency for the LSC)
1986	20,000 (calculated)
1986–1997	10,000 (estimated based on previous reported MDAs)
1998 – present	5,500

Note: Reporting limits were based on dose not bioassay results.

Table 5-8. Detection limits for total uranium bioassay[a].

Method	Period	MDA (µg/L)
No data	<1968	5 (based on state of technology)[b]
Fluorimetry	1968	5 (based on reported error)
Fluorimetry	1969 –1973	5 (estimated based on 1968 data)
Fluorimetry	1973 –1986	1.1 (based on lowest reported value in individual employee bioassay reports)
Fluorimetry	1975	0.7 (based on individual employee data
Fluorimetry	1986 –1987	3
Fluorimetry	1987–1990	10

CEP	1991–1992	12 (highest reported in previous years from Table 5-2)[c]
ICP-MS	1993 – Present	0.1

a. No reporting limits were available in the documents provided to the authors.
b. MDA of 5 µg/L is consistent with the value reported for bioassay conducted at nearby LLNL site (ORAUT 2005c) and the Hanford site (ORAUT 2004a).
c. Significantly higher than the years immediately following.

Table 5-9. Solubility type, fraction activity and particle size.

Facility	Nuclide	Solubility type	Particle size
TRL (Building 968)	^{3}H	Tritiated water	Not applicable
Building 913 Machine Shop	DU	Metallic form – unspecified solubility type but likely Type S	No data
Building 968	Uranium tritide	Metallic Type S; no data to show tritide is more soluble than metal	No data
Building 913	UO_2	Type S	No data

6.0 OCCUPATIONAL EXTERNAL DOSE

6.1 PURPOSE

The purpose of this section is to detail historical external dosimetry programs, systems, and practices at SNL/CA. This information may be used by dose reconstructors as needed to evaluate external occupational doses for EEOICPA claimants in terms of supplementing individual dose records for monitored SNL/CA workers, estimating respective missed doses, or estimating doses for unmonitored workers.

6.2 SCOPE

Historical documentation related to radiological protection programs at SNL/CA indicates that external dosimetry monitoring for workers and visitors at SNL/CA has been performed throughout the site's history. The information provided in this section draws on review of currently available records. The reivew for years prior to 1991 contains less information due to fewer available supporting documents. In the event further relevant documents are found for any period during the site's history, this section of the SPD will be revised accordingly.

6.3 INTRODUCTION

Historically, tritium exposure has been the primary radiological concern at SNL/CA (Ullrich 2003, DOE 2006). However, tritium exposure is not significant in terms of external dose. As noted in Section 2.3, the laboratory typically handled kg amounts of DU, typically in the form of alloyed metal components, g amounts of ^{3}H, and µCi quantities of other isotopes. The Radiography Facility (Building 923) also contained 100 Ci ^{192}Ir and ^{60}Co sources, a sealed ^{252}Cf source, and many other smaller sealed sources of activity ranging from one Ci to 500 mCi (see Section 2.3). According to a DOE Web site, uranium exposure potential at SNL/CA is extremely small and there is no potential for plutonium exposure (DOE 2006). Radioactive waste at SNL/CA is categorized as low-level. High-level or transuranic waste is not generated or stored at SNL/CA (SNL/NM / ES&H 1992). Table 2-1 summarizes the SNL/CA buildings of potential radiological interest.

With respect to radiation protection at SNL/CA, from 1956 to 1959 Lawrence Livermore National Laboratories (LLNL) performed external film dosimetry monitoring services for SNL/CA (SNL/CA 1958–1961). After that, a commercial vendor called Radiation Detection Company (RDC) provided film dosimetry services until the early 1970s (SNL/CA 1958–1961, SNL/CA 1961–1962, SNL/CA 1962, DeSelm 1965, SNL/CA 1964, Lovell 1966, RDC 1969). In 1962, there was a brief transfer of contracted film processing services from RDC to a similar provider called Tracerlab (SNL/CA 1962), but Tracerlab was quickly dropped due to poor performance and the contract with RDC was re-established. From about 1972 to 1988, SNL/CA external dosimetry was outsourced to the DOE Radiological and Environmental Sciences Laboratory (RESL) in Idaho Falls, ID (Wright 1993, Wallace 1988, Ormond 1986). In 1989, dosimetry services for SNL/CA were transferred to SNL headquarters in Albuquerque, New Mexico (SNL/NM).

In 1991, the dosimetry processing laboratory at SNL/NM became DOE Laboratory Accreditation Program (DOELAP)-accredited (Loesch 1991) as part of the overall plan to provide centralized, unified, and permanent dosimetry services for SNL/CA and other Sandia sites (Stanley 1991; SNL/NM /ES&H 1992; Ward et al. 1994). Most of the dosimetry information presented, described, and evaluated in this section of the SPD for years after 1991 was obtained at SNL/NM by the ORAU Team for the purpose of developing a SPD for that site (those records are directly applicable to the dosimetry program at SNL/CA from 1991 forward). Although the records currently available for this

SPD do not fully describe many technical details of dosimetry programs before DOELAP accreditation in 1991, they do provide an indication of the types of dosimeters used, exchange periods, and in most cases, types of radiation dose quantities that were measured and recorded.

6.4 DOSE RECONSTRUCTION PARAMETERS

6.4.1 Site Historical Administrative Practices

6.4.1.1 Administrative Practices Before 1989

Documents from early in SNL/CA's history indicate a policy of maintaining permanent dosimetry records (SNL/CA 1963). Between 1956 and 1959, most SNL/CA employees, contractors and visitors were required to wear dosimeter badges (SNL/CA 1958-1961, SNL/CA 1958-978). During this time, LLNL was performing dosimetry services for SNL/CA, with respective records management implemented manually by SNL/CA personnel. Records management continued to be administered by SNL/CA (division 8242-2) for the subsequent period in which RDC provided dosimetry services (1959 to about 1971). By 1965, discussions among SNL/NM management were taking place with respect to the need to badge all personnel on site versus badging only personnel with access to "exclusion zones" (areas where radiation fields were present). Eventually, the policy of badging all personnel was abandoned in favor of badging only individuals with potential to exceed certain exposure limits, which appears to have occurred in 1970, according to summarized annual reports of external exposures (SNL/CA 1958-1978). Through about 1972, dosimetry results were recorded manually on 4 by 6-in. cards or were stored as text file hard copies, all of which were transferred to SNL/NM in 1993 (Wright 1993). Electronic external dosimetry records for years between 1973 and 1988 were transferred to SNL/NM in 1989 (Wright 1993), apparently to become part of a master electronic records database management program called SANDOS. Since 1989, dosimetry records for all Sandia sites have been retained and managed by SNL/NM (Ward et al. 1994).

6.4.1.2 Administrative Practices After 1989

In 1989, the dosimetry program at SNL/CA was transferred to SNL/NM and has since been directed and managed by SNL/NM's Personnel Dosimetry Department 7715, with on-site oversight and implementation conducted by SNL/CA's Personnel Dosimetry Department 8541 (Ward et al. 1994). Dosimetry records generated by Department 7715 are entered into SANDOS. Some example personnel dose history output (SNL/NM 1995a) is shown in Attachment B (Figure B-1). Although dose units are not stated in SANDOS output, review of other documents (SNL/NM 1995a) suggests that they are reported in rem. There are specific protocols and training required for dosimetry data entry and records management (Ward et al. 1994).

Consistent with DOE Order 5480.11, Section 9.g. (DOE 1989), administrative external dose limits for SNL/CA personnel are as follows:

- 100 mrem (0.001 sievert) annual effective whole body dose equivalent
- 5 rem (0.05 sievert) annual dose equivalent to the skin
- 5 rem (0.05 sievert) annual dose equivalent to any extremity
- 1.5 rem (0.015 sievert) annual dose equivalent to the lens of the eye

Any site personnel with potential to receive annual doses exceeding these limits are required to wear personal dosimeters (Thompson et al. 1991; Ward et al. 1994). Monitored workers with interim dosimetry results indicating that annual doses could approach or exceed these limits may have work restrictions imposed. In 1997, a memo was issued (SNL/NM 1997) announcing an apparent change in dose limit reporting protocols under SNL's ALARA policy (Figure 6-1). These limits are similar to those listed above, but it is unclear whether they represent a combined total of both internal and external doses.

Attached are the monthly ALARA Radiation Dosimetry Reports. The reports list the occupational radiation dosimetry records for those employees whose annual dose equivalent exceeds 2, 6, and 10 percent of the limiting values specified in 10 CFR 835, Occupational Radiation Protection.

When an employee's annual dose equivalent exceeds the ALARA Report action levels, the individual's records will appear on subsequent monthly reports until the end of the current year. 10 CFR 835 limiting values and the ALARA Report action levels are given in the following table.

ORGAN	10 CFR 835 LIMITING VALUES (rem)	ALARA REPORT ACTION LEVELS (rem)		
		2%	6%	10%
Whole Body	5.0	0.1	0.3	0.5
Skin	50.0	1.0	3.0	5.0
Extremity	50.0	1.0	3.0	5.0
Lens of Eye	15.0	0.3	0.9	1.5

Figure 6-1. ALARA reporting action levels.

There are two dosimeter categories for monitoring personnel on site: routine field dosimeters and nonroutine field dosimeters. Routine field dosimeters are required (as specified above) for regular employees or contractors working at the site for extended periods, and results are reported on a quarterly basis. Nonroutine field dosimeters are issued to short-term temporary workers, site visitors, or personnel performing special radiation work outside the scope of their normal activities (i.e. performing nonroutine work tasks where the potential for additional or special types of exposures may exist).

While not made explicitly clear in Section 8.4.1 of SNL's External Dosimetry Program Manual (Ward et al. 1994), routine field dosimeter exchanges appear to occur on a monthly or quarterly basis, while nonroutine exchanges occur on a bi-weekly basis (for special radiation work cases, exchange periods may vary).

Before doses are calculated from gross field TLD readings, the amount of measured thermoluminescence due to background radiation is subtracted to obtain a net result due only to radiation from occupational exposure (Bradley et al. 1994, 1995; Walker 1996). Background thermoluminescence is the summation of "system background" and "environmental background." System background consists of thermoluminescence inherent in the instrumentation (e.g. noise due to photomultiplier tubes), and is determined from re-read values for daily measurements of calibration cards. Environmental background consists of accumulated exposures from cosmic and terrestrial

background sources, and is determined from trip control dosimeters that accompany each batch of field dosimeters to and from each SNL site. In addition to monitoring exposures during shipping, trip control dosimeters remain stored at daily check-in and check-out locations while at a given SNL site so that only site-specific background radiation is accumulated. The locations of badge storage areas for SNL/CA have not been identified.

Routine field dosimeters are assigned and issued by Department Managers and are subject to specific protocols for on site use, as well as subsequent handling and shipment to the Personnel Dosimetry Division in the New Mexico office for processing. Personnel issued routine field dosimeters are required to have documented training with respect to dosimeter use, as well as general radiation safety training.

Dosimeter shipping and handling quality control protocols include the use of special zippered envelopes for transport, dosimeter issue and return lists, chain-of-custody control documentation, trip control cards, express (overnight) shipping, and upon arrival at the NM Processing Center all field dosimeters are inspected for physical damage and card assignments are verified (Ward et al. 1994).

Nonroutine field dosimeters for temporary employees and visitors are issued by the Security Patrol Division at a specified site check in locations. The Security Inspector is responsible for insuring proper documentation of personnel being issued nonroutine field dosimeters, and for instruction in their use while on site. With the apparent exception of dosimeter exchange periods, protocols for nonroutine field dosimeter handling and shipment to the NM Processing Center are the same as those employed for routine field dosimeters.

Nonroutine field dosimeters for special radiation work cases are requested from Division 7715 and issued by Department Managers for any personnel involved in such work (Walker 1995). Respective examples of reporting for this type of nonroutine evaluation are provided in Attachment B (Figures B-2 and B-3).

Dosimetry results greater than 1 rem, questionable results, suspected misuse, processing difficulties, and non-returned field dosimeters will result in a formal investigation to resolve the issue as well as document and justify the dose equivalent assigned to the respective individual's dosimetry record. Radiation work restrictions may be imposed pending resolution of an investigation. Example documentation of a non-returned dosimeter investigation and assignment of missed dose are provided in Attachment B (Figures B-4 to B-6).

The overall performance of SNL's Personnel Radiation Dosimetry Program (PRDP) is monitored quarterly by a blind audit (Ward et al. 1994), but it is unclear as to who performs these audits.

In addition to personnel radiation dosimetry, SNL/CA conducts a Special Dosimetry Program, which includes work area monitoring (Ward et al. 1992; SNL/NM 1995b). These dosimeters are also considered "nonroutine" and respective exchange periods can vary depending on the intended monitoring purpose. Request forms and example results report for work area dosimeters are provided in Attachment B (Figures B-7 and B-8). Shipping and handling procedures are identical to those for personnel field dosimeters and the dose equivalent results are also recorded in SANDOS (SNL/NM 1995b).

6.4.2 Site Dosimetry Technology

The historical progression of dosimeter technologies used for external dose monitoring at SNL/CA is shown in Table 6-1. As broken down and discussed in the following sub-sections, site dosimetry

technologies have been partitioned into two historical categories (before 1989 and after 1989). This is because for the purposes of this SPD, more complete and detailed information is available to describe the dosimetry program after 1989.

Table 6-1. Documented dosimetry technologies and dose quantities.

Dosimeter Type	Years	Dosimetry Service Provider	Measured Quantities	Exchange Frequency	Compliance Dose Quantities
Two-element Beta/Photon film + NTA Neutron film (1, 2, 3, 4, 5)	1956 - 1959 (5)	LLNL	Photons, Beta, Neutron, Penetrating, Non-penetrating (1)	Monthly (5) or Quarterly (1)	(AEC manual, chpt 0524)
Two-element Beta/Photon film + NTA Neutron film (1, 2, 3, 4, 5, 12, 13)	1959 - 1971	RDC	Photons, Beta, Neutron, Penetrating, Non-penetrating (3)	Monthly (4, 11) or Quarterly (9, 11)	(AEC manual, chpt 0524)
2-chip TLD (one filter type only) (7)	1972 - 1982 (7)	RESL	No discrimination of different radiations (7); Photons, Beta, Neutron, Penetrating, Non-penetrating (8)	Semi-annual (7)	(14) ANSI N13/WG-2 (N324) 1978: Penetrating/non-penetrating; Photons: <0.3MeV, 0.3-10MeV; Betas: 0.2-2.0MeV; photon/beta mixtures
3-element Eberline TLD (7)	1982 - 1988 (6)	RESL	Photons, Beta, Neutron, Penetrating, Non-penetrating (7, 8)	Annual (6) or Semi-annual (7, 10)	(14) ANSI N13/WG-2 (N324) 1978: Penetrating/non-penetrating; Photons: <0.3MeV, 0.3-10MeV; Betas: 0.2-2.0MeV; photon/beta mixtures
Multi-element Harshaw TLD (separate special dosimeter for Neutrons)	1989 - 1990	SNL/NM	Shallow (Sh) Deep (Dp) Neutron (Nt)	Quarterly	Skin = Sh + Dp + Nt WB = Dp + Nt Extremities = Sh + Dp + Nt
Multi-element Harshaw TLD	1991-current	SNL/NM (DOELAP)	Shallow (Sh) Deep (Dp) Neutron (Nt)	Quarterly	Skin = Sh + Dp + Nt WB = Dp + Nt Extremities = Sh + Dp + Nt

LLNL = Lawrence Livermore National Laboratory
RDC = Radiation Detection Company
RESL = Radiological and Environmental Sciences Laboratory, DOE Idaho

(1) Dosimetry Printout 1957-1965
(2) Dosimetry Program Suppliers 1962
(3) Dosimetry Positive Exposures 1965
(4) Film Dosimetry Program 1964
(5) Miscellaneous Dosimetry Details 1958-1961
(6) External Dosimetry Trends 1989
(7) Dosimetry History 1993
(8) SNL/CA External Dosimetry '73-'87, Volumes 1&2 (UCI controlled dosimetry records)
(9) External Dosimetry Data 1961-62
(10) Radiation Dosimetry at SNLL: Past, Present and Future 1986 (RDSPPF1986)
(11) Miscellaneous Dosimetry Details 1965
(12) Film Badge Numbers 1966 (Lovell 1966)
(13) Dosimetry Report Examples 1969 (RDC 1969)
(14) 1983 Dosimetry Program Memo

6.4.2.1 Dosimetry Technology Before 1989

As suggested by certain documents cited in Table 6-1, dosimeters worn by SNL/CA personnel between 1956 and the early 1970s consisted of two-element Du Pont Type 554 beta/photon film along with a special nuclear track emulsion, type A film. Film badges consisted of four windows with one open and the others filtered with various densities of Pb, Cd, and Al (RDC 1963). Apparently, neutron film was exchanged on a monthly basis, while a quarterly exchange period was generally implemented for beta/photon film. There is a discrepancy in the available record as to exactly when SNL/CA discontinued using film dosimeters and began using the thermoluminescent dosimeter (TLD). *Dosimetry History 1993* (Wright 1993) indicates that this switch occurred in about 1966. Lovell 1966 indicates continued use of film in 1966, with plans to continue that use. Lovell (1983) estimates the use of TLDs began in about 1968. Radiation Detection Company (1969) indicates that they were still processing badges for SNL/CA as late as 1970, but it is not clear whether TLD technology was in use at the time. It is possible that the switch to TLD's coincided with a switch of dosimetry service providers from RDC to RESL in Idaho Falls, Idaho (Wright 1993), or that there was some overlap in usage of film and TLDs. Due to the uncertainty, it is favorable to the claimant, in cases where missed dose is estimated, to assume that film dosimeters were used at SNL/CA until about 1972.

As reported in Wright (1993), up until 1982, SNL/CA's TLD technology consisted of a 2-chip badge with only one filter type [30 mg/cm2 total filtration according to Wright (1981b)] and as such, could not discriminate between different radiation types or energies. However, the actual dosimetry records for this period report doses in penetrating and nonpenetrating dose categories, including beta, photon, and neutron radiations. If the TLD's during this period could not directly measure these different dose quantities, it is not clear if (or how) adjustments to recorded dose were made. Wright (1981b) suggests that nonpenetrating dose may have also been assigned as penetrating dose, but the document does not provide any indication of how different radiations, including neutron doses, were assessed. In 1982, SNL/CA switched to a three-element TLD from Eberline that apparently could better discriminate between different radiation types and provide directly measurable indicators of these various dose quantities. SNL/CA continued using the Eberline dosimeter until about 1989.

6.4.2.2 Dosimetry Technology After 1989

Since 1989, radiation dose monitoring at SNL/CA has been based primarily on the use of Harshaw TLD's. The Harshaw 8800 series TLD systems were the first to be DOELAP accredited at SNL/NM, and are the predominant systems used at all Sandia sites. Harshaw 4000 series TLD's are apparently utilized as well (Stanley et al. 1987; SNL/NM 1996a), though little information is currently available on the specific application and extent of use of this series. There is some evidence that the 4000 series may have been used, at least until 1996, for extremity dose monitoring (Walker 1997a). In 1997, SNL began using Harshaw/Bicron EXTRAD dosimeters for extremity monitoring. There is some indication that in addition to routine dosimeters for regular workers in the radiography facility, pocket dosimeters were used for nonroutine personnel accessing that facility (Lovell 1984a, 1984b).

Because the Harshaw 8800 system is a primary method for external personnel dosimetry monitoring at SNL/CA, it is pertinent to describe some basic technologies involved. Model 8801 and 8802 TLD cards consist of four thermoluminescence elements encapsulated between two thin sheets of Teflon, all of which is sandwiched between aluminum jackets. The jackets have four holes positioned over the thermoluminescence elements so that they can be heated by hot N_2 gas in the card reader. For operational efficiency, SNL color-codes TLD card edges according to use (e.g. calibration cards have green edges, quality control cards have red edges, field cards for worker monitoring have no added coloring).

The assembled dosimeter consists of the TLD card positioned inside a Model 8812 card holder. The cardholder front has radiation modifying filters to evaluate radiation and dose equivalent quantities as shown in Figure 6-2. For simplicity, the manufacturer refers to the entire assembly as a Model 8812 dosimeter, while SNL refers to this assembly as the SNL dosimeter.

The Harshaw 8800 card reader is an automated system in which up to 1400 cards can be loaded at a time and read automatically. Barcode ID information printed on each card is automatically recorded by the card reader before heating. Precisely controlled heating causes the TLD elements to give off light in proportion to the amount radiation they have received. The light produced is converted to an electrical signal by a photomultiplier tube. The relative strength of the electrical signal is measured in units of charge (nanocoulombs) creating "glow curves" that are then analyzed with respect to card reader calibration parameters. Non-contact heating of the cards with hot nitrogen gas improves dosimeter re-use, durability, and improves glow-curve reproducibility. Raw data from the card reader is acquired, analyzed, and stored with TLD Radiation Evaluation and Management System software resident on a desktop computer. Card reader results are converted into dose equivalent by a complex algorithm developed specifically for the Harshaw 8800/8812 system and SNL sites.

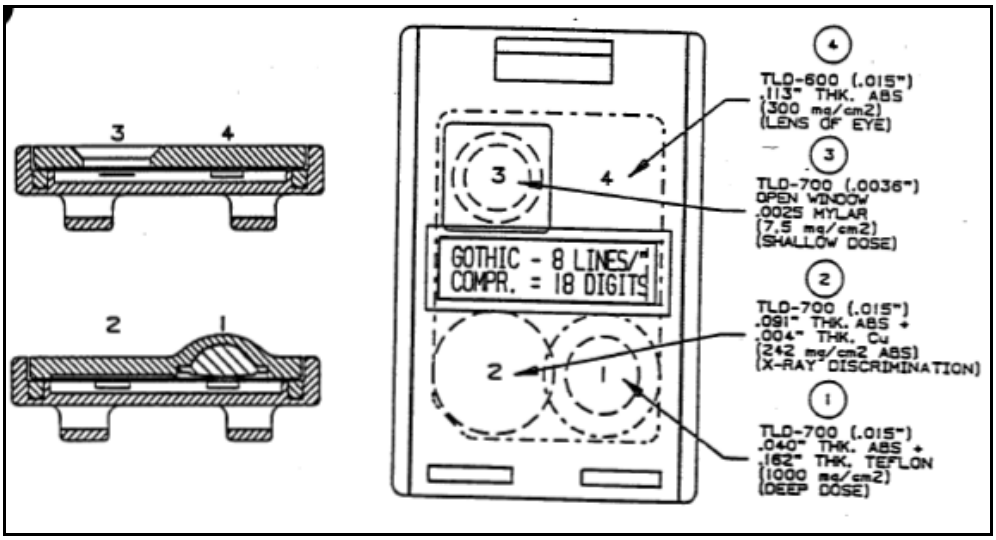

Figure 6-2. Harshaw Model 8812 dosimeter assembly, element specifications, and respective radiation, and dose equivalent quantities measured by Rhea and Bradley (1990).

Along with explicit protocols for card reader calibration, DOELAP accreditation is an on-going process that includes repeated algorithm validation and blind audits as part of an overall data quality assurance program. Calibration and algorithm protocols are described in more detail in subsequent sections of this report.

6.4.3 Calibration

6.4.3.1 Calibration for Dosimeter Technologies Before 1989

Calibrations for film dosimeters processed by RDC in the 1960s involved exposing film in SNL/CA badges to known Co-60 gamma and X-ray fields (effective X-ray energies were 35 keV and 90 keV) based on prior calibration data for RDC badges (RDC 1963). SNL/CA badges had Al, Cd, and Pb filters, each of which was cross-calibrated against similarly filtered RDC badges in parallel runs. Currently, no information is available regarding the calibration of TLD badges used by SNL/CA between 1971 and 1988 (the period in which RESL performed dosimetry services).

6.4.3.2 Calibration for Dosimeter Technologies After 1989

Harshaw Model 8800 TLD Card readers are setup and maintained in accordance with manufacturer recommendations. Figure 6-3 shows a calibration procedure flowchart (Bradley et al. 1995) along with an example calibration checklist, glow curves, and calibration results output (SNL/NM 1996b).

CARD READER CALIBRATION CHECKLIST

READER #: 2 Day: 1/3/96 Julian Day: 003

Performed By: [illegible]

INITIAL CHECKS	OK	Changed/Updated
N_2 Supply	✓	
UPS Function	✓	
Equipment On	✓	
Printer Ribbon	✓	
Printer Paper	✓	
Date and Time	✓	
Cleaned Clear Optical Filters:	Yes ✓	No

ELECTRONICS QC Passed:	Yes ✓	No		
QC CARDS INCLUDED	Yes ✓	No	# 10	
CALIBRATE CARD READER				

TTP # 1
Function "Anneal Cards" Group File # 0039620
Irradiate Cards @ 840 gU Start @ 07:59:26 end @ 08:31:26
Function "Calibrate Reader" Group File # 0039600

SCREENS PRINTS AND TLDREMS REPORTS	Screen Prints	TLDREMS Reports	Review
Electronics QC			
Re-set Parameters	✓		
Results	✓		
Data Acquisition			
TTP #1	✓		
Acquisition Set-up	✓		
Read Cards	✓		
Cal Card Anneal w/ glow curves	✓	✓	
Read Cards		✓	
Cal Card Read w/ glow curves			
Reader Calibration Report	✓	✓	
Plot QC data (prior to upload)	✓		

COMMENTS:

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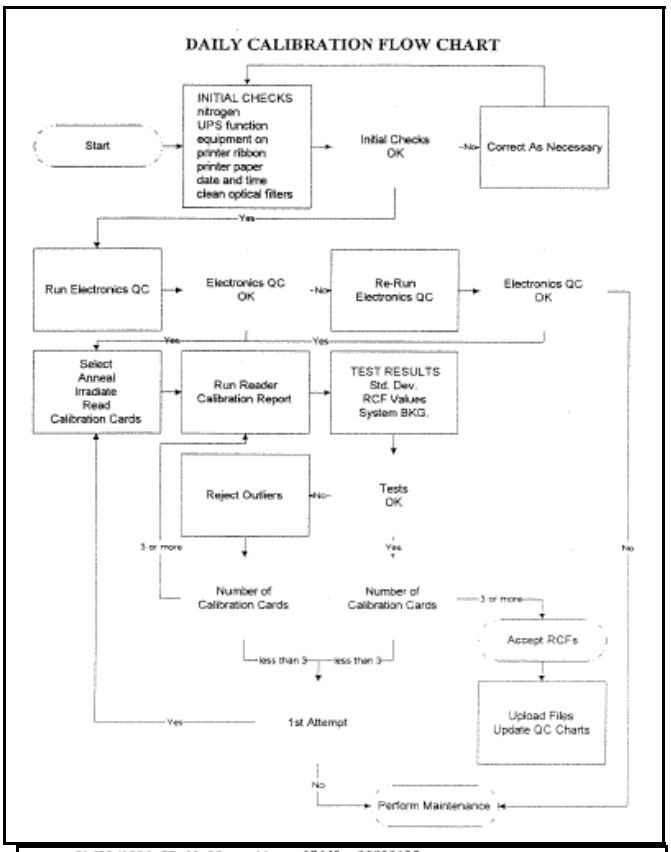
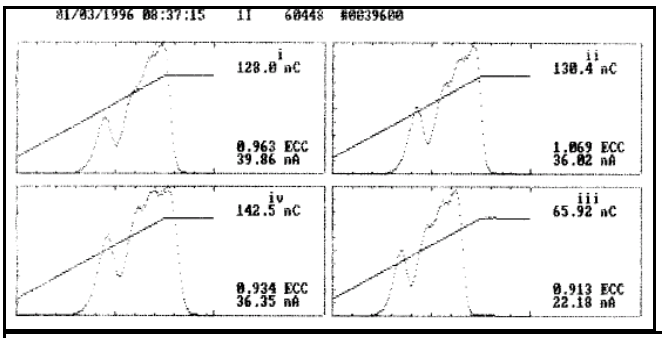
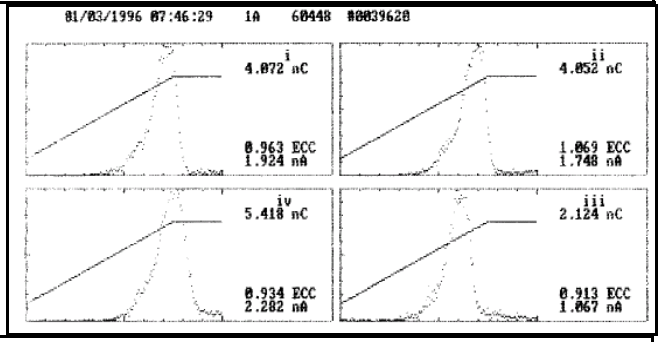




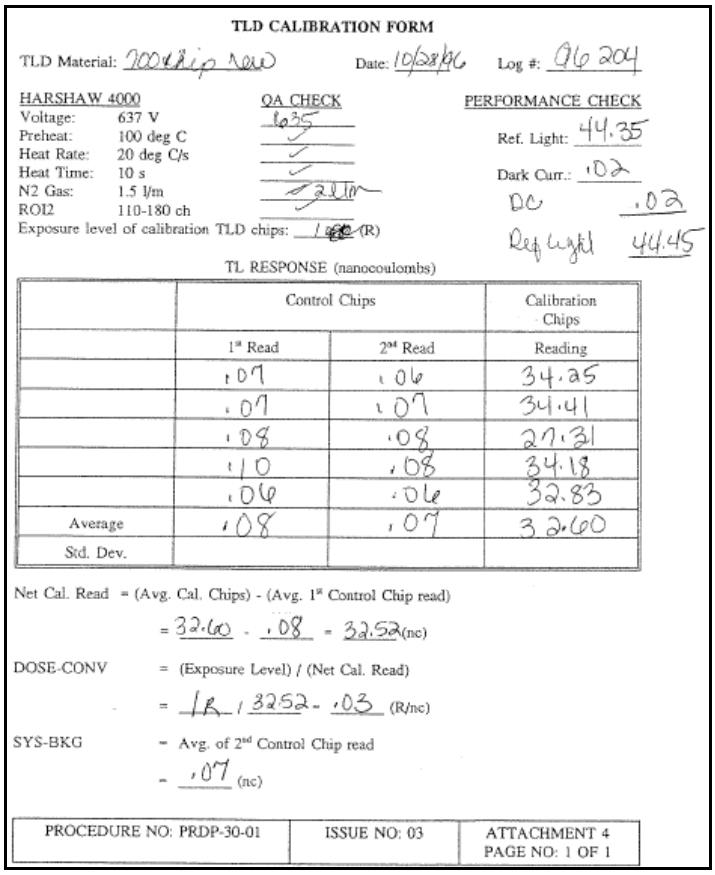


Reader Summary	Samples	(i)	(ii)	(iii)	(iv)	Units
New Calibration Factor (RCF)		0.3767	0.3814	0.1913	0.4151	nC/gU
Old Calibration Factor (RCF)		0.3775	0.3858	0.1917	0.4141	nC/gU
Mean Reading	5	128.08	129.70	65.05	141.16	nC
% Std. Dev.		1.064	1.090	1.607	1.072	
Ref. Light Mean	7	377.88	257.56	378.72	315.59	nC
Ref. Light % Std. Dev.		0.4779	0.5019	0.3915	1.045	
PMT Noise Mean	7	0.0783	0.0467	0.1043	0.1047	nC
PMT Noise % Std. Dev.		18.53	11.31	17.87	17.76	

Figure 6-3. Example Harshaw 8800 card reader calibration procedures, forms, analysis, and output.

Specific details of the Harshaw 8800/8812 system calibration procedures can be found in Bradley et al. (1993) and Rhea and Bradley (1990). Procedures appear different for calibration of the Harshaw 4000 system (SNL/NM 1996a) and an example calibration form is shown in Figure 6-4. No official manuals or procedural documents have been found concerning Harshaw 4000 dosimetry system

calibration or system applications, nor has any similar documentation been found regarding the Harshaw/Bicron EXTRAD system.



TLD CALIBRATION FORM

TLD Material: 200 chip new Date: 10/28/96 Log #: 96 204

HARSHAW 4000 — QA CHECK

Voltage:	637 V	635
Preheat:	100 deg C	✓
Heat Rate:	20 deg C/s	✓
Heat Time:	10 s	✓
N2 Gas:	1.5 l/m	✓ 2 l/m
ROI2	110-180 ch	✓

Exposure level of calibration TLD chips: 1 (R)

PERFORMANCE CHECK

Ref. Light: 44.35

Dark Curr.: .02

DC .02

Ref Light 44.45

TL RESPONSE (nanocoulombs)

	Control Chips		Calibration Chips
	1st Read	2nd Read	Reading
	.07	.06	34.25
	.07	.07	34.41
	.08	.08	27.31
	.10	.08	34.18
	.06	.06	32.83
Average	.08	.07	32.60
Std. Dev.			

Net Cal. Read = (Avg. Cal. Chips) - (Avg. 1st Control Chip read)

= 32.60 - .08 = 32.52 (nc)

DOSE-CONV = (Exposure Level) / (Net Cal. Read)

= 1R / 32.52 - .03 (R/nc)

SYS-BKG = Avg. of 2nd Control Chip read

= .07 (nc)

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Figure 6-4. Example calibration form for the Harshaw 4000 dosimetry system.

6.5 DOSE CALCULATION AND REPORTING

6.5.1 Dose Calculation and Reporting Before 1989

As shown in Table 6-1, reported doses for SNL/CA employees, contractors, and visitors before 1989 include penetrating and nonpenetrating categories for beta, photon, and neutron radiations. For the period in which film was used for measuring these quantities, it is assumed that the degree of film darkening under the various windows (filtered to different degrees) was used to estimate doses within these various reporting categories by comparison against calibration curves developed for each batch of film. No information is currently available with respect to minimum detection limits (MDLs), adjustments to recorded dose, uncertainty or bias, or adjustments to recorded dose during this period of film dosimeter use.

As mentioned previously, the 2-chip, single filter TLD dosimeters in use from about 1972 to 1982 apparently could not directly measure each of the various dose quantities reported in the record for this period. No definitive information has been located with respect to adjustments to recorded dose,

nor has any information been found regarding dose calculation algorithms, system MDLs, or uncertainty/bias doses during this period.

Like the earlier period in which 2-chip TLD's were used at SNL/CA and processed by RESL in Idaho, there is no documentation regarding dose calculation algorithms, MDLs, adjustments to recorded dose, or uncertainty/bias for the 3-element Eberline TLD badges used from 1982 to 1988.

6.5.2 Dose Calculation and Reporting After 1989

Doses to personnel and work area TLD cards are calculated per SNL/NM procedure PRDP-14-01 (SNL/NM 1991). Algorithms used for calculating dose equivalents are specific to each particular dosimeter system used. Both shallow (0.007 cm) and deep (1.0 cm) dose equivalents are measured, calculated, and reported. Although dose to lens of the eye is not measured directly, it can be calculated indirectly from routine dosimetry results. Due to the complexity of this calculation, however, SNL does not routinely calculate dose to lens of the eye, but instead uses an annual cumulative shallow dose equivalent of 1.5 rem as a benchmark value that triggers respective manual calculation and reporting of results (Walker 1997b). No information has been found on extremity calculation and reporting.

There are specific circumstances and respective protocols for making adjustments to an individual's recorded dose or estimating doses when reliable measurements are not available (Potter et al. 1993; SNL/NM 1995a). Some primary examples are shown in Table 6-2. Calculation or database programming errors are also included, but don't necessarily involve an adjustment. For example, in 1997 a SANDOS programming oversight was discovered that had resulted in incorrect background subtractions since 1992, resulting in slightly overestimated respective doses (Walker 1997c). The programming correction was made when discovered, but no adjustments to the affected prior records were implemented since the slight overestimations were considered conservative.

Table 6-2. Circumstances and protocols for dose adjustment or estimation.

Calculated Dose Level	Dose Assignment
Calculated dose* < 10 mrem for deep or neutron dose equivalent.	Report 0 rem
Calculated dose* < 35 mrem for deep or neutron dose equivalent.	Set shallow dose equivalent equal to deep dose equivalent
Calculated neutron dose* > 100 mrem (not applicable to work area dosimeters)	Contact user to verify neutron code, reassess and adjust doses if justified
Missed Dose (non-returned or lost dosimeter)	Estimate dose based on prior history, dose to other personnel, or reconstruct based on work area dosimetry results
Suspected problem with analysis (damaged card, processing difficulties, etc.)	Estimate dose based on prior history, dose to other personnel, or reconstruct based on work area dosimetry results
Calculation or related software programming mistakes	Correct calculation or programming errors and adjust records as warranted and justified

*Effective dose equivalent calculated per PRDP-14-01

The angular dependence of the Harshaw 8812 TLD was studied by SNL as part of the DOELAP accreditation process (Friedman et al. 1991). A batch of 8812 TLD's was sent to Battelle Pacific Northwest Laboratory (PNL), which maintains an National Institute of Standards and Technology (NIST)-traceable irradiation and measurement quality assurance program, to assess the angular dependence of respective dosimeter readings for various exposure geometries. The results (Figure 6-5) led to a conclusion that the horizontal and vertical angular dependence of the TLD system used by SNL are acceptable and "comparable to or better than results for modern dosimetry systems."

Horizontal Orientation — Harshaw 8800/8812 TLD System

Category/Source	Test Depth	-85°	-60°	-30°	0°	+30°	+60°	+85°
IIIA (M150)	Deep	0.932	1.124	1.208	1.234	1.197	1.072	0.653
	Shallow	0.729	1.102	1.214	1.257	1.224	1.130	0.787
IIIA (M30)	Deep	0.881	0.907	1.133	1.197	1.046	0.801	0.200
	Shallow	0.268	0.741	1.170	1.211	1.158	0.760	0.162
IV	Deep	0.987	0.985	1.005	1.019	0.998	0.954	0.578
	Shallow	0.809	0.916	0.962	0.980	0.975	0.944	0.929
VA ($^{90}Sr/^{90}Y$)	Shallow	0.015	0.316	1.037	1.112	0.982	0.156	0.109
VI (moderated)	Deep	0.477	0.889	1.185	1.209	1.091	0.821	0.401
VI (unmoderated)	Deep	0.052	0.609	1.095	1.097	1.051	0.556	0.274

Vertical Orientation — Harshaw 8800/8812 TLD System

Category/Source	Test Depth	-85°	-60°	-30°	0°	+30°	+60°	+85°
IIIA (M150)	Deep	0.924	1.126	1.221	1.253	1.196	1.072	0.663
	Shallow	0.495	1.049	1.209	1.262	1.229	1.154	0.839
IIIA (M30)	Deep	0.589	0.899	1.150	1.183	1.098	0.804	0.291
	Shallow	0.142	0.705	1.141	1.146	1.113	0.810	0.178
IV	Deep	1.007	1.008	1.011	1.004	0.999	0.946	0.824
	Shallow	0.787	0.870	0.947	0.971	0.987	0.945	0.959
VA ($^{90}Sr/^{90}Y$)	Shallow	0.026	0.171	0.976	1.086	1.063	0.336	0.030
VI (moderated)	Deep	0.158	0.799	1.119	1.205	1.191	0.839	0.400
VI (unmoderated)	Deep	0.167	0.058	1.047	1.074	1.073	0.572	0.394

Figure 6-5. Angular dependence testing results for the Harshaw 8800/8812 dosimetry system. Results in each category are normalized (apparently to the average result from control cards exposed under the normal, perpendicular geometry).

The lower limits of detection (LLDs) for the Harshaw Model 8802 dosimeter were evaluated by SNL in 1997. Both monthly and quarterly exchange periods were tested within various DOELAP exposure categories and the results are shown in Table 6-3. An earlier LLD study, conducted in 1990 for the Model 8801 dosimeter, showed shallow dose equivalent LLD of about 31 mrem. The only difference between the Model 8802 and 8801 dosimeters is that the 8802 has a slightly thicker shallow chip.

Because the shallow dose LLD for the newer 8802 cards is 10 mrem or less across all exposure categories, SNL uses 10-mrem LLD value in SANDOS for shallow dose equivalent.

Table 6-3. Monthly and quarterly LLDs for the Harshaw Model 8802 dosimeter card (adapted from Walker et al. 1997).

Exposure Category	Radiation Type(s)	Test Depth	Monthly LLD (rem)	Quarterly LLD (rem)
I	Accident X-rays	Deep	0.003	0.004
II	Accident Gammas	Deep	0.003	0.005
IIIA	General X-rays	Shallow	0.004	0.008
		Deep	0.004	0.006
IV	Cs-137 Gammas	Shallow	0.003	0.007
		Deep	0.003	0.004
VA	Beta Particles	Shallow	0.003	0.007
VI	Moderated Neutrons	Deep	0.003	0.005
VII (mixtures)				
	III + IV	Shallow	0.005	0.010
		Deep	0.004	0.006
	III + VA	Shallow	0.004	0.008
		Deep	0.005	0.008
	III + VI	Deep	0.003	0.004
	IV + VA	Shallow	0.004	0.008
		Deep	0.004	0.006
	IV + VI	Deep	0.003	0.004

6.5.3 Exposure Energy Spectra

Dose reconstruction under the NIOSH program (NIOSH 2002) requires estimates of exposure percentages within specific energy bands for each type of radiation as follows:

Photons

- < 30 keV
- 30 – 250 keV
- > 250 keV

Betas

- < 15 keV
- > 15 keV

Neutrons

- < 10 keV
- 10 – 100 keV
- 100 keV – 2 MeV
- 2 – 20 MeV
- > 20 MeV

For external exposures, betas with energies < 15 keV are not applicable. Table 6-4 provides estimates of percentages of radiation energies within each relevant category, broken down by process within each facility. Most of these estimates were obtained from SNL/CA site personnel. Other sources and assumptions are provided as applicable in the table footnotes. The Tritium Research Laboratory is not listed because the primary radionuclide was tritium with beta energies < 15 keV.

Table 6-4. Selection of radiation energies and percentages.

Process description	Radiation type	Energy selection (keV)	Percentage[a]
Weapons Laboratory Facility Complex (buildings 910, 912, 913, 914, 916, 918) Years: 1958 - 1998			
Test/repair of neutron and X-ray detectors (neutron and X-ray generators)	Photon	>250	100
	Neutron	10-100	5
		100-2,000	5
		2,000-20,000	90
Wet machining of DU	Beta	>15	100
	Photon	30-250	50[b]
		>250	50[b]
Radiography for weapons components	Beta	>15	100
	Photon	30-250	50[b,c]
		>250	50[b,c]
Radiography for materials science studies (X-ray diffraction operations later moved to building 941)	Beta	>15	100
	Photon	<30	40
		30-250	30
		>250	30
3H storage studies	Photon	30-250	50[b,d]
		>250	50[b,d]
Ion beam analysis of materials	Photon	30-250	30
		>250	70
Radiflo leak tests	Beta	>15	100
	Photon	30-250	10
		>250	90
Radiography (building 923) Years: unknown – early 1990's			
Radiography using X-rays, gamma rays, neutrons, alpha and beta particles	Beta	>15	100
	Photon	<30	5
		30-250	45
		>250	50
	Neutron	10-100	5
		100-2,000	70
		2,000-20,000	25
Micro and Nano Technologies Laboratories (buildings 941, 942, and 943) Years: unknown – current			
Radiography for materials science studies (radiological materials in building 941 only)	Beta	>15	100
	Photon	<30	40
		30-250	30
		>250	30
Explosives and Environmental Testing Complex (buildings 955, 956, 966, 972, 974, 976, 977, 978, 979, 981, 983) Years: 1958 – current			
Environmental testing of mock-up weapons and components (DU)	Beta	>15	100
	Photon	30-250	50
		>250	50
Storage Facilities (buildings 921, 927, 961, 982) Years: unknown – current			
Storage and packaging of waste materials	Beta	>15	100
	Photon	30-250	70
		>250	30

a. Estimated primarily by site personnel.
b. Based on favorable to claimant default assumptions provided for DU in ORAUT 2006b.
c. Assumes radiography primarily associated with weapons mock-ups and DU.
d. Assumes only external exposures would be associated with depleted uranium beds in tritium storage facility.

6.5.3.1 Neutron Dose Conversion Factors

As described in ORAUT-OTIB-0055 (ORAUT 2006c), adjustments to neutron dose are necessary to account for changes in quality factors between historical and current scientific guidance. Using the methodology indicated in ORAUT-OTIB-0055, adjustment factors were determined for the various energy groups at the two SNL/CA facilities in which neutron exposures were possible. Table 6-5 shows multiplier values to use for dose reconstruction.

Table 6-5. Neutron dose conversion factors for applicable SNL/CA facilities.

Facility	Neutron energy intervals	NCRP 38 Quality factor	ICRP 60 weighting factor	Dose fraction[a]	Corrected dose equivalent multiplier[b]
Weapons laboratory complex	10 - 100 keV	5.38	10	0.05	0.093
	0.1 - 2.0 MeV	10.49	20	0.05	0.095
	2.0 - 20.0 MeV	7.56	10	0.90	1.190
Radiography	10 - 100 keV	5.38	10	0.05	0.093
	0.1 - 2.0 MeV	10.49	20	0.70	1.335
	2.0 - 20.0 MeV	7.56	10	0.25	0.331

a. From Table 6-4.
b. Multiply reported dose by these factors to determine the corrected neutron dose equivalent for each applicable neutron energy interval.

6.5.4 Missed Dose

Given the lack of documentation regarding MDLs for dosimeter systems for the period prior to 1989, it is necessary to estimate dosimeter system MDLs and missed dose. During this early period, it is reasonable to assume that MDLs for the various systems were similar to contemporary technologies used at other AEC or DOE facilities during corresponding periods. Maximum annual missed doses are estimated using exchange periods and MDL/2 as recommended by NIOSH (NIOSH 2002). After 1989, MDL information presented in the previous section is used to estimate missed dose using exchange periods and MDL/2. Table 6-6 lists documented or estimated MDLs and maximum missed doses for each dosimetry technology used by SNL/CA.

Table 6-6. Estimated maximum annual missed photon, beta, and neutron dose.

Period of use	Dosimeter	MDL (mrem)	Exchange frequency	Maximum annual missed dose (mrem)[a]
1956 - 1959	Two-element Beta/Photon film	30[b]	Monthly; Quarterly	60; 20
1959 - 1971	Two-element Beta/Photon film	30[b]	Monthly; Quarterly	60; 20
1956 - 1971	Neutrons (NTA film)	50[b]	Monthly; Quarterly	60; 20
1972 - 1982	2-chip TLD	20[c]	Semi-annual	15
1982 - 1988	3-chip Eberline TLD	20[c]	Annual; Semi-annual	8; 15
1972 - 1988	Neutrons (2-chip & 3-chip TLD systems)[d]	20[c]	Annual; Semi-annual	8; 15
1989 - 1990	Multi-element Harshaw TLD	10	Quarterly	20
1991-current	Multi-element Harshaw TLD	10	Quarterly	20
1989-current	Neutrons (Harshaw TLD systems)	5	Quarterly	10

a. Maximum annual missed dose calculated using N x MDL/2 from NIOSH (2002).
b. Estimated MDL based on contemporary film systems of similar sensitivity (ORAUT 2004b and ORAUT 2005d).

c. Estimated MDL based on contemporary TLD systems of the period (ORAUT 2005d, ORAUT 2006d, and ORAUT 2006e).
d. Although neutron doses are reported for the 2-chip system, no documentation of the neutron dosimetry methodology employed has been found.

6.5.5 Organ Dose Conversion Factors

During the period in which film dosimeters were used at SNL/CA, dosimeters were calibrated in units of roentgen (R). No information is available regarding calibration of TLDs for the period between 1972 and 1988, but it is favorable to claimants to assume TLDs were calibrated in units of exposure (R). After dosimetry services were transferred to SNL/NM to gain DOELAP accreditation and consolidate all SNL dosimetry programs, the Personal Dose Equivalent [Hp(10)] should be used. Table 6-7 shows dose units to use for organ dose conversion factors.

Table 6-7. Dose units for organ dose conversion factors.

Years	Photon dose units for use with organ dose conversion factors
1956 - 1959	R
1959 - 1971	R
1972 - 1982	R
1982 - 1988	R
1989 - 1990	Hp(10)
1991-current	Hp(10)

6.5.6 Uncertainty

Given the lack of specific technical information obtained with respect to dosimetry systems for much of SNL/CA's history, it is necessary to estimate respective measurement uncertainty based on reported values for contemporary systems in use at other facilities. Table 6-8 shows estimates based on the technical basis document for the Hanford Site (ORAUT 2006d), with additional values for NTA film adapted from ORAUT 2006e. Some general analogies can be drawn between Hanford and SNL/CA in terms of dosimetry technologies employed during various time periods of interest.

Table 6-8. Bias and uncertainty estimates (adopted from ORAUT 2006d).

Dosimeter	Period of use	Bias magnitude and range		Uncertainty factors	
		Overall bias[a]	Range in bias	Systematic[b]	Random[c]
Two-element film	1956 - 1971	1.02	0.86 – 1.12	1.1	1.4
Two-chip TLD	1972 – 1982	1.12	1.04 – 1.16	1.05	1.2
Three-chip TLD	1982 - 1988	1.01	0.95 – 1.05	1.05	1.2
Multi-element TLD	1989 - current	1.00	0.95 – 1.05	1.05	1.2
NTA film[d]	1956 - 1971	1.5	0.5 – 1.5	1.5	

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

7.0 ATTRIBUTIONS AND ANNOTATIONS

Where appropriate in this SPD, bracketed callouts have been inserted to indicate information, conclusions, and recommendations provided to assist in the process of worker dose reconstruction. These callouts are listed here in the Attributions and Annotations section with information provided to identify the source and justifications for each associated item. Conventional references are listed in the next section, linking data, quotations, and other information to documents available for review on the NIOSH Project computer network.

[1] Thomas, Elyse. ORAU Team. Principal Medical Dosimetrist. October 2006. The X-ray record form included in claim files has a place for the technician to record the projection, mA, kVp, distance, and time. For PA chests, the mA is almost always listed as 5, the distance is listed as 72", and the kVp is usually around 70. Exposure times are not usually listed. It is assumed that the person completing this part of the X-ray record form was recording the mAs, not the mA, primarily since radiographic machines are not designed to operate at such low mA settings, and also because time settings were not recorded. Five mAs would not be unusual for a PA chest exposure.

[2] Lopez, Theresa. ORAU Team. Senior Toxicologist. September 2006. Lumber spine X-rays were performed at hire from 1956 through 1971 as evidenced by medical records in claim files.

[3] Lopez Theresa. ORAU Team. Senior Toxicologist. September 2006. Per review of X-ray records and at the direction of ORAU in comments dated October 2006, the X-ray machines have been assumed to be single-phase.

[4] Lopez Theresa. ORAU Team. Senior Toxicologist. September 2006. Lumbar spine X-rays were performed at hire from 1956 through 1971 as evidenced by medical records in claim files.

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GLOSSARY

alpha particles
Positively charged particles of discrete energies emitted by certain radioactive materials; alpha particles usually expend their energy in short distances and will not usually penetrate the outer layer of skin; they are a significant hazard only when taken into the body where their energy is absorbed by tissues.

becquerel (Bq)
A special unit of activity. One becquerel equals one nuclear transition per second.

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

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

depleted uranium (DU)
A very dense by-product of the enrichment of natural uranium, which contains about 99.8% ^{238}U, 0.2% ^{235}U and 0.001% ^{234}U by mass.

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

dosimetry
The science of assessing absorbed dose, dose equivalent, effective dose equivalent, etc., from external or internal sources of radiation.

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

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

film
In general, a "film packet" that contains one or more pieces of film in a light-tight wrapping. When developed, the film has an image caused by radiation that can be measured using an optical densitometer.

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

fission
The splitting of a heavy atomic nucleus, accompanied by the release of energy.

fissionable
Material capable of undergoing fission.

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

isotope
Elements having the same atomic number but different atomic weights; identical chemically but having different physical and nuclear properties.

neutron
A basic particle that is electrically neutral weighing nearly the same as the hydrogen atom.

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

operating area
Designation of major onsite operational work areas.

radiation
Alpha, beta, neutron, and photon radiation.

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

radionuclide
A radioactive isotope of an element, distinguished by atomic number, atomic weight, and energy state.

rem
A unit of dose equivalent equal to the product of the number of rad absorbed and the quality factor.

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

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

ATTACHMENT A
SUMMARY OF ACTIVITIES AT SITE BUILDINGS
Page 1 of 3

Building Number	Building Use	Radiological Activities	Dates of Use	Nuclides	Comments
904	Auditorium	None	1980s–present		
905	Offices	None	1980s–present		
906	Combustion Laboratory Research	None	1980s–present		
907	Equipment room	None	1980s–present		
910	Weapons Laboratory Facility	Neutron Detection			Used to design, assemble, test, calibrate and repair neutron detectors
911	Administration	None			
912	Administration, computers	None	1958–present		
913	Test assembly, machine shop, plating, MetallographyTritium HVAHs	Tritium storage studies, U machining	1976–1998	H-3, U-238	
914	Test laboratories	Radiflow, NDT, U machining,		Kr-85, beta sources, U-238	No radiological work for ~15 yr
915	New building, office space	None			
916	Mostly light chemistry laboratories, Previously a warehouse, Ar gloveboxes	e-microscopes, accelerators, various small sources, H-2 storage tests		700 keV, 1 MeV, U-238	Sealed sources
917	Cylinder storage				
918	Raw stock	U-machining		U-238	Deconstructed in the late 1980s
920	Office space	None	1977–present		
921	Decommissioned in 1980	Radiological material storage and decontamination of NTS test units		U-238, trace Pu-239 and MFPs	
922	Office space	None			
923	Radiography Laboratory	Radiography operations; X-ray machines		Co-60, Ir-192 X-ray machines. Cf-252,	Converted to records storage in early 1990s
924 Mo	Mobile; Counting Laboratory				HP laboratory moved to Building. 973
925	Health services	None			

ATTACHMENT A
SUMMARY OF ACTIVITIES AT SITE BUILDINGS
Page 2 of 3

Building Number	Building Use	Radiological Activities	Dates of Use	Nuclides	Comments
927	Radiological Material Storage	Warehouse and Vault		U-238, natural thorium	
928	Shipping & receiving	None			
929	Office space	None			
940	Office space	None			
941	Light mechanical, electrical, chemical laboratories	Radiography		X-ray U-238, beta sources	Sealed sources
942	Light mechanical, electrical, chemical laboratories	None			
943	Plating operations				
955	Environmental Test Facility	Env. Testing of various Test units		U-238 metal	
956	Vibration Test Facility	Environmental Testing of various Test units		U-238 metal	
960	Offices				
961	Radiological and Mixed Waste Storage and packaging	Storage & packaging		H-3, U-238	
9611	Chemical waste storage	None			
963	Maintenance Facilities	None			
964	Security	None			
966	High Pressure Gas Dynamics Test Facility	None			
967	Office	None			
968	TRL	Tritium research	1976–1989	H-3, U-238 beds	Decommissioned in 1998, converted to biotech
969	Former TRL waste storage	Radiological waste storage for site	1976–1989	H-3, U-238	Converted to shop and storage when TRL decommissioned
970	Welding shop				
972	Centrifuge	Test package centrifuge, Penetrator studies		U-238	
973	ES&H laboratories				
974	Explosives test tanks	Neutron generator tests		H-3	
976	Four test cells for High Press work	H-2 storage tests		U-238	
978	Flight test unit testing	Mass properties tests		U-238	

ATTACHMENT A
SUMMARY OF ACTIVITIES AT SITE BUILDINGS
Page 3 of 3

Building Number	Building Use	Radiological Activities	Dates of Use	Nuclides	Comments
979	H-2 storage, research and development			U-238	Three Ar gloveboxes, two contaminated machines
983	Test cells; Flight Test Assembly			U-238	
ESA	Magazines; Explosives Storage			H-3	

Sources: DOE (1982); Wright (1981, 2006).

ATTACHMENT B
DOSIMETRY PROGRAM ADMINISTRATIVE DOCUMENTATION EXAMPLES
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DOSIMETRY PROGRAM ADMINISTRATIVE DOCUMENTATION EXAMPLES

Date: 11/15/95 ********* PRIVATE *********

Name.....: [redacted] Contact Org.....: 08613 Statu

Dates of Coverage		TEDE	Skin	Extrem.	Eyes	Deep	Shallow	Neutron
Fourth Quarter	1989	0.000	0.000			0.000	0.000	
Total for Year	1989	0.000	0.000			0.000	0.000	
First Quarter	1990	0.000	0.000			0.000	0.000	
Second Quarter	1990	0.000	0.000			0.000	0.000	
Third Quarter	1990	0.000	0.000			0.000	0.000	
Fourth Quarter	1990	0.000	0.000			0.000	0.000	0.000
Total for Year	1990	0.000	0.000			0.000	0.000	0.000
First Quarter	1991	0.000	0.000			0.000	0.000	0.000
Second Quarter	1991	0.000	0.000			0.000	0.000	0.000
Third Quarter	1991	0.000	0.000			0.000	0.000	0.000
Fourth Quarter	1991	0.000	0.011			0.000	0.011	0.000
Total for Year	1991	0.000	0.011			0.000	0.011	0.000
First Quarter	1992	0.000	0.000			0.000	0.000	0.000
Second Quarter	1992	0.010	0.010			0.010	0.010	0.000
Third Quarter	1992	0.000	0.000			0.000	0.000	0.000
Fourth Quarter	1992	0.000	0.000			0.000	0.000	0.000
Total for Year	1992	0.010	0.010			0.010	0.010	0.000
First Quarter	1993	0.000	0.000			0.000	0.000	0.000
Second Quarter	1993	0.000	0.000			0.000	0.000	0.000
Third Quarter	1993	0.000	0.000			0.000	0.000	0.000
Fourth Quarter	1993	0.000	0.000			0.000	0.000	0.000
Total for Year	1993	0.000	0.000			0.000	0.000	0.000
First Quarter	1994	0.000	0.000			0.000	0.000	0.000
Second Quarter	1994	0.000	0.000			0.000	0.000	0.000
Third Quarter	1994	0.000	0.000			0.000	0.000	0.000
Fourth Quarter	1994	0.000	0.000			0.000	0.000	0.000
Total for Year	1994	0.000	0.000			0.000	0.000	0.000
First Quarter	1995	0.000	0.000			0.000	0.000	0.000
Second Quarter	1995	0.000	0.000			0.000	0.000	0.000
Total for Year	1995	0.000	0.000			0.000	0.000	0.000

******** End of Report ********

Note: Year 0000 is for all years prior to 1987.

Figure B-1. Example external dosimetry history records (output from SANDOS).

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DOSIMETRY PROGRAM ADMINISTRATIVE DOCUMENTATION EXAMPLES
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NON-ROUTINE PERSONNEL DOSIMETER EVALUATION

Page No. 1 VERIMOD05

REASON Possible exposure

☐ High Readout ☒ Other (specify) Possible Exposure (to NEUTRONS)

☒ Special Evaluation Requested by: Tom Laiche (Name) 6/1/95 (Date)

USER INFORMATION

Name: [redacted] Social Security Number: [redacted]

Neutron Code: 90 Assigned Org. (at time of issue): 7714 Mail Stop:

LOCATION - Room: Worksite Code: SA

Additional location information, if available: QUARTERLY ISSUED DOSIMETRY USED FOR NON-ROUTINE WORK. NEUTRON CODE STILL VALID DUE TO SHIELDING

ASSIGNED Use Dates: From 6/1/95 to 6/30/95 ACTUAL Use Dates: From 6/1/95 to 6/1/95

DOSIMETER INFORMATION

Dosimeter Number: [redacted] File Name: [redacted] Annual Date: 5/25/95

Read Date: 6/1/95 Background Days:

READOUT (gu) - Element #1: 2.902 Element #2: 3.595 Element #3: 1.646 Element #4: 38.307

BACKGROUND - (if applicable)

BACKGROUND RATE (gu/day)	ENVIRONMENTAL BKG (gu)	SYSTEM BKG (gu)	TOTAL BKG (gu)
Element #1	Element #1 0.980	Element #1 ~~0.980~~ 0.645	Element #1
Element #2	Element #2 0.980	Element #2	Element #2
Element #3	Element #3 1.120	Element #3	Element #3
Element #4	Element #4 1.540	Element #4	Element #4

TRIP CONTROL INFORMATION - (if applicable) N/A

File Name:

Annual Date:

Read Date:

GROSS AVERAGE READOUT (gu)

Element #1 Element #2

Element #3 Element #4

Figure B-2. Example nonroutine dosimeter evaluation form for special radiation work case (potential neutron exposures).

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DOSIMETRY PROGRAM ADMINISTRATIVE DOCUMENTATION EXAMPLES
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Page No. 2 VER040203

CHECKLIST

TO DO	DONE	TO DO	DONE
☐ Additional QC Cards Read		☒ Copy Glow Curves of Card Read, ~~Re-Read~~	SW
☒ Visually Inspect Card(s)	SW	☒ Copy TLD Card Reader Daily Calibration	SW
☐ Reread Card(s)		☐ Other (specify)	
☐ Generate ECC(s) (if directed by Dosimetry Staff Leader)			

Comments: CARD & GLOW CURVE INSPECTIONS WERE SATISFACTORY

Work Performed By: S. WALKER (Name) 6/2/95 (Date)

DOSE - (rem)

DEEP	SHALLOW	NEUTRON
0.000	0.000	0.000

Comments: NEUTRON CODE NOT CHANGED SINCE INDIVIDUAL WAS STANDING WELL OFF FROM 252Cf source, behind a moderated shield.

Dose Calculated By: SCOTTIE WALKER (Name) 6/2/95 (Date)

CONTACT REQUESTOR - (if applicable)

TOM LAICHE (Name) 7714 (Org.) (Mailstop) 6/5/95 (Date) 0905 (Time)

Phone Number Where Requestor Can Be Reached or Message Left:

DOSIMETRY PROJECT LEADER ACTIONS

Concurrence/~~Non-Concurrence~~ Comments: 7714 personnel informed of the need to request special dosimetry when working on jobs outside their normal scope.

SCOTTIE WALKER (Dosimetry Project Leader) Scottie Walker (Signature of Dosimetry Project Leader) 6/5/95 (Date)

ACTION TAKEN - Shaded area to be completed by 7715 personnel ONLY

Brief Explanation (if necessary): No neutron code change was warranted since the individual was located behind a water moderator.

Dose Assignment Entered into SANDOS By: S. WALKER IC Code(s) (if any): IC 247 Date: 6/6/95

Figure B-3. Example nonroutine dosimeter evaluation form for special radiation work case (potential neutron exposures).

ATTACHMENT B
DOSIMETRY PROGRAM ADMINISTRATIVE DOCUMENTATION EXAMPLES
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Report Date: 11/15/95

PRIVATE
Non-Returned Dosimeter Investigation Report

RETURN TO: MS 065

CUSTOMER DATA

Dosimeter No. [redacted] Assigned To: [redacted] SS No.: [redacted]

Dosimeter Type: TLD Whole Body Issue Date: 07/01/95 Due Date: 09/30/95 Current Org.: 08613

Assigned Org During Issue Period in Question: 08613 Period of Coverage from 07/01/95 to 09/30/95

Company Name (if individual is not a Sandia employee): JOHNSON CONTROLS

REASON FOR INVESTIGATIO

LOST DOSIMETER

INFORMATION ONLY

EMPLOYEE'S PREVIOUS RADIATION EXPOSURE HISTORY

Dose for Prior Period: 04/01/95 to 06/30/95, is 0.0 deep 0.0 shallow 0.0 neutron rem

Dose for Current Year-to-Date: 1995, is 0.0 deep 0.0 shallow 0.0 neutron rem

Dose for Previous Year: 1994, is 0.0 deep 0.0 shallow 0.0 neutron rem

THE EXPOSURE OF PERSONNEL WORKING WITH EMPLOYEE DURING THE PERIOD IN QUESTION

Name of Employee	Social Security No.	Dates of Exposure	Dose (rem)
			____ deep ____ shallow ____ neutron
			____ deep ____ shallow ____ neutron
			____ deep ____ shallow ____ neutron

DOSE ESTIMATE

The BodyDose Estimate for the period 07/01/95 to 09/30/95, is 0.0 deep 0.0 shallow 0.0 neutron rem

The Estimate is Based on: [X] Previous History [ ] Exposure of Other Personnel

[ ] Other _____

DOSE ESTIMATE APPROVALS

If dose estimate is not acceptable, list reason(s) on a separate sheet and return all correspondence for further consideration.

I have reviewed the dose estimate for the period 07/01/95 to 09/30/95 and agreed th

0.0 deep 0.0 shallow 0.0 neutron (rem) is the Body dose that should be recorded for this period.

- Employee Signature: [redacted] Org: 8613 Date: 11-29-95
- Department Manager Signature: (Department or Above) Org: 8613 Date: 11-29-95
- Health Physicist Signature: N/A (Required only if assigned dose is 0.100 mrem or greater) Org: ____ Date: ____
- Radiation Dosimetry Signature: Org: 7715 Date: 12/4/95

ACTION TAKEN (Needed only to be completed by 771 only following dose estimate approval) Assigned Badge #: 007142

Reviewed by Dosimetry Project Leader: Remarks Code: Entry Date: 12/7/95 Entered By: Filed By:

Figure B-4. Example unreturned dosimeter investigation report.

ATTACHMENT B
DOSIMETRY PROGRAM ADMINISTRATIVE DOCUMENTATION EXAMPLES
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Sandia National Laboratories
Albuquerque, New Mexico 87185

date: 11/15/95

to: Manager, 08613

from: Radiation Protection Measurements Dept. 7715, MS 0651

subject: Non-Returned Dosimeter Questionnaire - Follow-Up

reference: [redacted] 08613 SS#: [redacted]

Dosimeter Use Period: from 07/01/95 to 09/30/95

Assigned Org. at Issue: 08613 Current Org.: 08613

Dosimeter No.: [redacted] Dosimeter Due Date: 10/14/95

Thank you for your cooperation on the Non-Returned Dosimeter Questionnaire. On the reverse side is the follow-up External Dosimetry Investigation Report. This report is based, in part, on the information you provided on the Non-Returned Dosimeter Questionnaire and is intended to facilitate your investigation. It is the responsibility of the Supervisor to complete the investigation, obtain the required signatures (including the appropriate Health Physicist), and return the completed report to the Dosimetry Distribution Center, Department 7715. Refer to the listing below to obtain the appropriate Health Physicist signature. If the investigation report is not returned within two weeks, a second notice will be generated and a copy sent to the next level of management. In addition, failure to resolve investigations may result in the issuance of a work restriction for the individual.
If you agree with the dose estimate, please obtain all signatures and return the completed report to the Dosimetry Distribution Center, Department 7715 within two weeks so that we may update our records.

If either you or the employee disagree with the dose estimate or if you have additional information, please note this information on the investigation report and return it to Radiation Dosimetry Division 7715, within two weeks. Following additional investigation, another report will be sent for your signature.

If you have any questions regarding this Non-Returned Dosimeter Investigation Report, call 4-7725.

VER921204

Copy to:
07715 Dosimetry File

AREA HEALTH PHYSICISTS

Area I	Richard Stump	844-5943
Area II & IV	Martin Chee	845-7602
Area III & V	Tom Laiche	845-3271
Livermore	Donn Wright	(510) 294-2615
NTS	Jim Metcalf	(702) 295-

Figure B-5. Example unreturned dosimeter questionnaire follow-up letter.

ATTACHMENT B
DOSIMETRY PROGRAM ADMINISTRATIVE DOCUMENTATION EXAMPLES
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PART IV If a dosimeter cannot be located an estimated dose value must be determined and assigned to the individual's dosimetry history file. The purpose of this section is to establish the method used for making the estimation.

Indicate the most suitable method for dose estimate by checking one of the boxes below and providing any additional information as required:

☒ Use individual prior dose history.
☐ Use doses assigned to one or more of the individual's co-workers. List appropriate co-workers below.

NAME | SOCIAL SECURITY NUMBER

☐ Other (specify area survey, time motion, etc.). Give supporting information on a continuation sheet...

Individual Assigned Dosimeter (print name) | Signature | 10/22/95 Date

J.G. Peterson
Department Manager (print name) | Signature | 10/24/95 Date

ESSENTIAL CODES

WORKSITE	CODE
Sandia National Lab in NM	SA
Sandia National Lab in CA	SL
Pantex Facility in TX	PX
Nevada Test Site	NT
Tonopah Test Range in NV	TT
QA Rep Valley Fargo, PA	VF
White Sands Missile Range	WS
TRW in CA	TR
WIPP Site in NM	WP
Lawrence Livermore Lab in CA	LL
Edwards Air Force Base	ED
McCarran Airport in Las Vegas	MC
Vandenburg Air Force Base	VN
Kauai, Hawaii	KA
Whole Body Badge (for Travel)	XB

FACILITY	CODE
Accelerator	10
Maintenance/Support	40
Reactor	50
Research, General	61
Research, Fusion	62
Waste Management	70
Weapons	80
Other	99

NEUTRON ENERGY	CODE
14 MeV	00
2.5 MeV	10
PuBe, AmBe	20
Cf-252	30
Lightly Moderated Fission Spectrum	60
Reactor Leakage Spectrum	70
Neutron Not Required	71
Thermal	90

OCCUPATION	CODE
Supervisor/Management	110
Engineer	160
Scientist	170
Health Physicist	184
Other Professional	200
Doctor/Nurse	260
Engineering Technician	370
Radiation Technician	383
Other Technician	390
Administrative Support	450
Security Inspector	513
Janitor	524
Repair/Construction	660
Shop Support	780
Transport	840
Other	990

Figure B-6. Example portion of unreturned dosimeter questionnaire showing various codes used by the Personnel Dosimetry Division.

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WORK AREA RADIATION DOSIMETRY REQUEST FORM

WORK AREA RADIATION DOSIMETRY

The information supplied on this form and on the accompanying 3.5" diskette will be used to process your work area dosimeters and to produce a written report which will be sent to the individual indicated in Section III below. Dosimeter placement information should be recorded on the reverse side of this document and on the accompanying 3.5" diskette. Return this form along with the dosimeters and the 3.5" diskette to Radiation Protection Measurements Department 7715.

SECTION I - Individual(s) responsible for the placement and/or retrieval of the dosimeters.

PLACEMENT Name: _____ (Last / First / Middle Initial) Org. No.: _____ Phone No.: _____

RETRIEVAL Name: _____ (Last / First / Middle Initial) Org. No.: _____ Phone No.: _____

SECTION II - Dosimeter placement and source information (see continuation on reverse side).

Facility Name: _____ Building/Room No.: _____

_____ Tech Area No. (1-5): _____

USE PERIOD: Date dosimeters placed: ___/___/___ (Mon / Day / Yr) Date dosimeters pulled: ___/___/___ (Mon / Day / Yr)

NUMBER OF: Dosimeters placed in area _____ Dosimeters retrieved at end of monitoring period _____

SOURCE INFORMATION: High Voltage _____ Operating Current _____
No. of Shots _____ Total Energy Generated _____
Source No. _____ Isotope _____
Other _____

Equipment manufacturer: _____ Model & Serial No.: _____

Neutron Energy Code: _____

AVERAGE NEUTRON ENERGY	CODE
14 (MeV)	00
2.5 (MeV)	10
Pulse, Audio	20
Cf-252 (Bare)	30
Lightly Moderated Fission Spectrum	60
Reactor Leakage Spectrum	70
Neutron Dosimetry Not Required	71
Thermal	90

SECTION III - Reporting requirements.

☐ Deep Dose ☐ Shallow Dose ☐ Neutron Dose ☐ Dose per Shot

SEND REPORT TO: _____ (Last / First / Middle Initial) Org. No.: _____ Phone No.: _____

FOR DOSIMETRY ORGANIZATION USE ONLY

Log. No.: _____ Entered By: _____ (Initials) ___/___/___ (Mo / Day / Yr) Filed By: _____ (Initials) ___/___/___ (Mon / Day / Yr)

Figure B-7. Work Area Radiation Dosimetry Request Form.

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EXAMPLE REPORT

SANDIA NATIONAL LABORATORIES
HEALTH INSTRUMENTATION DIVISION
SPECIAL DOSIMETRY

Report Date: 02/20/95
Log Number: 95003
Customer: [redacted] MS 1093
Facility: SCO
Field Cycle: 01/05/95 to 01/05/95
Neutron Code: 90
Source Information: BLDG 963 ANNEX: 9 SHOTS
NEUTRON DOSE RESULTS ASSUME DOSIMETERS MOUNTED ON HYDROGENOUS BACKSCATTERER

Number of Shots: 9

Dosimetry Performed By: D. C. Ward, Dept. 7715

The referenced dosimetry was performed using thermoluminescent dosimeters (TLDs). The results are reported in dose equivalent units of rem for Deep (1.0 cm) and Shallow (0.007 cm) tissue depths. The values have had the background dose subtracted. The average dose per shot is also reported.

The statistical or random errors associated with the Deep and Shallow results are less than the larger of +/- 0.005 rem or +/- 10% at the 95% confidence level. The systematic errors are less than +/- 10% and are primarily due to photon energy dependence. The Lower Limit of Detection (LLD) for the deep and shallow dose is 0.010 rem at the 95% confidence level. The dosimetry calibration is performed daily and has traceability to NIST.

The statistical or random errors associated with the neutron results are less than the larger of +/- 0.005 rem or +/- 20% at the 95% confidence level. The systematic errors can be as large as a factor of two (2) and are primarily due to uncertainties in the selection of the neutron fluence and dose conversion factors. The Lower Limit of Detection (LLD) for the neutron dose is a function of the neutron spectrum and ranges from 0.010 to 0.120 rem for low energy to 14 MeV neutrons. The neutron calibration is performed on a phantom and has traceability to NIST.

If there are any questions regarding this information, please call the Dosimetry Office at 844-7197.

Figure B-8. Example work area radiation dosimetry results report.