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PUBLICATION RECORD

EFFECTIVE DATE	REVISION NUMBER	DESCRIPTION
11/07/2003	00	New technical basis document for the Idaho National Engineering and Environmental Laboratory – Site Description. First approved issue. Initiated by Norman D. Rohrig.
07/28/2004	01	Corrects dates of operation in Section 2.4.11 and adds OGC required paragraphs. Approved issue of Revision 01. Initiated by Norman D. Rohrig.
07/29/2005	02	Adds additional descriptive material on ICPP in response to union outreach comments. Incorporates formal internal and NIOSH review comments. Approved issue of Revision 02. Retraining is not required. Initiated by Norman D. Rohrig.
08/17/2007	03	Approved Revision 03 initiated to add Attributions and Annotations section. Added references. Incorporates formal internal and NIOSH review comments. Training required: As determined by the Task Manager. Initiated by Jo Ann M. Jenkins.
08/02/2010	04	Revision initiated to combine INL and ANL-W Site Descriptions. Incorporates formal internal and NIOSH review comments. Training required: As determined by the Objective Manager. Initiated by JoAnn M. Jenkins.

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ACRONYMS AND ABBREVIATIONS

630-A	High Temperature Marine Propulsion Reactor
710	Fast Spectrum Refractory Metals Reactor
AEC	U.S. Atomic Energy Commission
AFSR	Argonne Fast Source Reactor
ANL-W	Argonne National Laboratory-West
ANP	aircraft nuclear propulsion
ARA	Army (later Auxiliary) Reactor Area
ARMF	Advanced Reactivity Measurement Facility
ARVFS	Army Reentry Vehicle Facility Site
ATR	Advanced Test Reactor
ATRC	Advanced Test Reactor Critical (facility)
BORAX	Boiling-Water Reactor Experiment
BWR	boiling-water reactor
CE	Critical Experiment
CET	Critical Experiment Tank
CFA	Central Facilities Area
C.F.R	Code of Federal Regulations
Ci	curie
CPP	Chemical Processing Plant
CRCE	Cavity Reactor Critical Experiment
CTF	Core Test Facility
D&D	decontamination and decommissioning
DOE	U.S. Department of Energy
DOELAP	DOE Laboratory Accreditation Program
DOL	U.S. Department of Labor
DOT	U.S. Department of Transportation
DU	depleted uranium
EBR	Experimental Breeder Reactor
EBOR	Experimental Beryllium Oxide Reactor
EEOICPA	Energy Employees Occupational Illness Compensation Program Act of 2000
EFS	Experimental Field Station
EOCR	Experimental Organic-Cooled Reactor
ETR	Engineering Test Reactor
ETRC	Engineering Test Reactor Critical (facility)
FAST	Fluorinel Dissolution Process and Fuel Storage (facility)
FDF	Fluorinel Dissolution Facility
FEBT	Fuel Element Burn Test
FMF	Fuel Manufacturing Facility
FPF	Fuel Processing Facility
FPR	fuel processing restoration
FSR	Fast Source Reactor (same as AFSR)
ft	foot
g	gram
gal	gallon

GCRE	Gas-Cooled Reactor Experiment
HEU	highly enriched uranium
HFEF	Hot Fuel Examination Facility
HLLW	high-level liquid waste
HOTCE	Hot Critical Experiment
HPIL	Health Physics Instrument Laboratory
HPP	Hot Pilot Plant (later Headend Processing Plant)
hr	hour
HTRE	Heat Transfer Reactor Experiment
ICDF	INL Comprehensive Environmental Response, Compensation, and Liability Act Disposal Facility
ICPP	Idaho Chemical Processing Plant
IET	Initial Engine Test
IFSF	Irradiated Fuel Storage Facility
in.	inch
INL	Idaho National Laboratory
INTEC	Idaho Nuclear Technology and Engineering Complex
ISFSI	independent spent fuel storage installation
ITR	Intrepid Technology and Resources
keV	kiloelectron-volt, 1,000 electron-volts
kg	kilogram
kVp	kilovolts-peak, applied kilovoltage
kW	kilowatt
kWt	kilowatt thermal
L	liter
LOCA	loss-of-coolant accident
LOFT	Loss-of-Fluid Test
LPTF	Low Power Test Facility
m	meter
MAPs	mixed activation products
mCi	millicurie
MeV	megaelectron-volt, 1 million electron-volts
MFC	Material and Fuels Complex
MFPs	mixed fission products
mi	mile
min	minute
mL	milliliter
ML-1	Mobile Low-Power Reactor No. 1
mR	milliroentgen
mrad	millirad
mrem	millirem
MTR	Materials Test Reactor
MW	megawatt
MWt	megawatt-thermal
nCi	nanocurie
NIOSH	National Institute for Occupational Safety and Health
No.	number

NRAD	Neutron Radiography Facility
NRC	Nuclear Regulatory Commission
NRF	Naval Reactors Facility
OMRE	Organic-Moderated Reactor Experiment
ORAUT	Oak Ridge Associated Universities Team
PBF	Power Burst Facility
POC	probability of causation
psi	pounds per square inch
psig	pounds per square inch gauge
Pt.	part
PWR	pressurized-water reactor
R	roentgen
RAL	Remote Analytical Laboratory
RaLa	radioactive lanthanum
RCRA	Resource Recovery and Conservation Act
RFP	Rocky Flats Plant
RMF	Reactivity Measurement Facility
RML	Radiation Measurements Laboratory
RPSSA	Radioactive Parts Service and Storage Area
RWMC	Radioactive Waste Management Complex
s	second
SL-1	Stationary Low-Power Reactor No. 1
SMC	Specific Manufacturing Capability
SNAP	System for Nuclear Auxiliary Power
SNAPTRAN	System for Nuclear Auxiliary Power Transients
SPERT	Special Power Excursion Reactor Test
SRDB Ref ID	Site Research Database Reference Identification (number)
STPF	Shield Test Pool Facility, also known as SUSIE
TAN	Test Area North
TBD	technical basis document
TMI	Three Mile Island
TRA	Test Reactor Area
TREAT	Transient Reactor Experiment and Test (facility)
TSA	Transuranic Storage Area
TRU	transuranic
TSF	Technical Support Facility
U.S.C.	United States Code
W	watt
Wt	watt-thermal
WCF	Waste Calcining Facility
WIPP	Waste Isolation Pilot Plant
WRRTF	Water Reactor Research Test Facility
yd	yard
yr	year

ZPR	Zero Power Reactor
ZPPR	Zero Power Plutonium (later Physics) Reactor
µCi	microcurie
°	degrees angle
°C	degrees Celsius
°F	degrees Fahrenheit
§	section or sections

2.1 INTRODUCTION

Technical basis documents (TBDs) and site profile documents are not official determinations made by the National Institute for Occupational Safety and Health (NIOSH) but are rather general working documents that provide 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, 42 C.F.R. Pt. 82) define "performance of duty" for DOE employees with a covered cancer or restrict the "duty" to nuclear weapons work (NIOSH 2007a).

The statute also 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 excludes Naval Nuclear Propulsion Facilities from being covered under the Act, 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 occupational radiation exposures are considered valid for inclusion in a 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 (NIOSH 2007a):

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

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

2.1.1 Purpose

The purpose of this TBD is to provide a description of the Idaho National Laboratory (INL) and the Argonne National Laboratory-West (ANL-W) that contains technical basis information to evaluate the total individual occupational dose for claimants under EEOICPA.

INL and ANL-W played a major role in early reactor research and development. The site has operated 52 reactors and conducted fuel handling, fuel reprocessing, and radioactive waste storage and disposal since it began operations in 1949. The purpose of this TBD is to assist in the evaluation of worker dose from INL and ANL-W processes using the methodologies in OCAS-IG-001, *External Dose Reconstruction Implementation Guideline* (NIOSH 2007b), and OCAS-IG-002, *Internal Dose Reconstruction Implementation Guideline* (NIOSH 2002).

2.1.2 Scope

This TBD provides supporting technical data with assumptions that are favorable to claimants to evaluate the total INL and ANL-W occupational dose that can be reasonably associated with worker radiation exposure covered under EEOICPA. This section describes the facilities, processes, and historical information in relation to worker internal and external exposures for use when actual monitoring data might be unavailable. *Proving the Principle*, *A History of the Idaho National Engineering and Environmental Laboratory, 1949-1999* (Stacy 2000) was used as a resource in the development of facility and process descriptions.

Section 2.1.3 provides general information about the site, and Sections 2.2 through 2.14 describe the site facilities and processes. Attributions and annotations, indicated by bracketed callouts and used to identify the source, justification, or clarification of the associated information, are presented in Section 2.15.

2.1.3 General Site Information

In 1949, the U.S. Atomic Energy Commission (AEC) established the National Reactor Testing Station in Idaho as a Federal reservation to build, test, and operate nuclear reactors. The site constructed and operated a variety of support facilities and equipment. In February 1951, AEC opened the Idaho Division of Argonne National Laboratory (which is now ANL-W) within the boundaries of the site. In 1974, the site became the Idaho National Engineering Laboratory, which in 1997 became the Idaho National Engineering and Environmental Laboratory. On February 1, 2005, the site became INL and combined the operations of the Idaho National Engineering and Environmental Laboratory, ANL-W, and the Idaho Cleanup Project, which is working on closure of inactive portions of the site. (For convenience, this TBD refers to the site as INL.)

INL is an 890-mi^2 reservation with a maximum distance of about 39 mi from north to south and 36 mi from east to west. It is 30 to 60 mi west of Idaho Falls, Idaho. Major site-related research facilities and offices are in Idaho Falls. The site is on the Snake River Plain in southeastern Idaho at an elevation of about 5,000 ft and is above the Snake River Plain aquifer.

INL is unique among DOE facilities because it is a large complex site with many independent technical areas, contractors, goals, and missions. Table 2-1 lists the prime contractors and the years during which they operated the site. The prime contractor provided some services to the other contractors and operated most of the facilities. Some of the specific technical areas were operated in part by other contractors during the tenure of the prime contractor. Table 2-2 lists those facilities and technical areas and the years during which contractors other than the prime contractor operated them.

Table 2-1. INL prime site contractors (Stacy 2000, p. 272).

Operating years	Prime contractor
1950–1966	Phillips Petroleum Company
1966–1972	Idaho Nuclear Corporation (a joint subsidiary of Allied Chemical Corporation, Aerojet General Corporation, and Phillips Petroleum Company)
1972–1976	Aerojet Nuclear Corporation
1976–1994	EG&G Idaho
1994–1999	Lockheed Martin Idaho Technologies Company
1999–2004	Bechtel BWXT Idaho
2005–Present	Battelle Energy Alliance for INL CH2M Washington Group for the Idaho Cleanup Project

Table 2-2. Other operating facility technical area contractors (Stacy 2000, p. 272).

Argonne National Laboratory-West	
1949–2004	University of Chicago
Advanced Mixed Waste Treatment Project	
2003-present	British Nuclear Fuel Limited
Idaho Chemical Processing Plant (later Idaho Nuclear Technology and Engineering Center)	
1950–1953	American Cyanamid Company
1953–1966	Phillips Petroleum Company
1966–1971	Idaho Nuclear Corporation
1971–1979	Allied Chemical Corporation
1979–1984	Exxon Nuclear Idaho Company
1984–1994	Westinghouse Idaho Nuclear Company
Aircraft Nuclear Propulsion Program	
1952–1961	General Electric Company
Specific Manufacturing Capability	
1983–1986	Exxon Nuclear Idaho Company
1986–1991	Rockwell Idaho National Engineering Laboratory
1991–1994	Babcock & Wilcox Idaho

Dosimetry services at INL (including ANL-W) were unique among DOE facilities from 1949 through 1988 in that DOE Idaho Operations Office personnel operated and provided internal and external dosimetry services. The dosimetry branch provided and analyzed external dosimetry badges, counted workers in the whole-body counter, and analyzed bioassay samples. DOE also provided portable radiation survey instruments and maintained and calibrated them. Personnel at the DOE Radiological and Environmental Sciences Laboratory in Building CF-690 in Central Facilities Area (CFA) maintained exposure histories of personnel based on dosimetry records, including bioassay data. With the advent of the DOE Laboratory Accreditation Program (DOELAP), DOE transferred dosimetry responsibility to the prime contractor on January 2, 1989, to eliminate a conflict of interest.

Figure 2-1 shows the location of each facility or technical area that is discussed in this TBD. The following sections describe each facility and its processes, with the exception of the Naval Reactors Facility (NRF), a naval propulsion facility that is exempt from EEOICPA requirements. The facilities include ANL-W [which operated the Experimental Breeder Reactor No. 2 (EBR-II), Transient Reactor Experiment and Test (TREAT) Facility, and Zero Power Plutonium (later Physics) Reactor (ZPPR)], Army (later Auxiliary) Reactor Area (ARA), CFA, Idaho Chemical Processing Plant (ICPP) [now the Idaho Nuclear Technology and Engineering Complex (INTEC)], EBR-I, Boiling-Water Reactor Experiment (BORAX), Radioactive Waste Management Complex (RWMC), Special Power Excursion Reactor Test (SPERT) area, Grid III [the test grid where the Fuel Element Burn Tests (FEBTs) A and

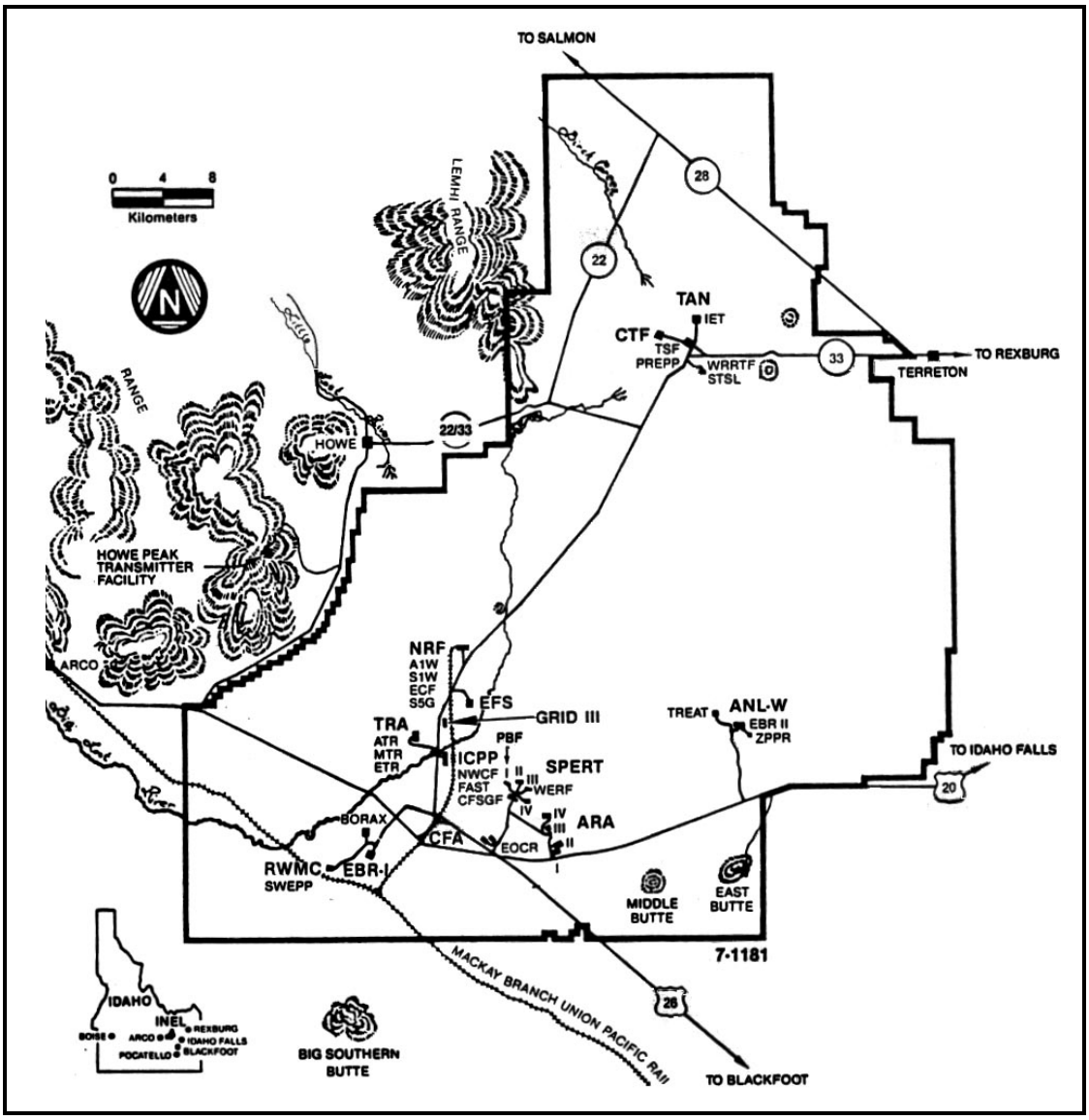

Figure 2-1. Idaho National Laboratory (Hoff, Chew, and Rope 1986, p. 40]).

B occurred], Test Reactor Area (TRA), the Experimental Field Station (EFS), NRF, Test Area North (TAN) [where the Initial Engine Tests (IETs) occurred], and the Core Test Facility (CTF) at TAN.

Personnel who worked at INL and ANL-W in designated radiological areas were typically required to wear state-of-the-art dosimetry (film badges, thermoluminescent dosimeters, and pocket ionization chambers), respiratory protection, anticontamination clothing, etc. [1]. Facilities and radiological areas were monitored by remote and portable remote area monitors, continuous air monitors, and air samplers. Portable units were used for work where fixed units were not available [2].

In cases where airborne radioactivity might have been present or internal exposure was possible, respirators were provided to prevent or reduce internal exposure [3].

Engineered systems were incorporated as practicable to minimize the potential for airborne radioactivity and direct radiation exposure. Bioassay programs were instituted to monitor and assess potential internal exposures. Environmental monitoring systems were placed around the site at

multiple locations to measure direct radiation, fallout, effluent discharges, and releases and to monitor and control exposures to onsite personnel and to members of the public. Each facility has had film badges or thermoluminescent dosimeters in specific building areas and around perimeter fences to measure direct environmental radiation accumulation at the location as a check on source terms in site locations [4].

In addition to the nuclear reactor experiments at TRA (Section 2.7), there have been other such experiments at INL. Table 2-3 lists common radionuclides for reactors.

Table 2-3. Radionuclides of concern for all reactors (INEEL 2001).

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

2.2 TEST AREA NORTH

TAN is 30 mi northeast of the CFA. General Electric Company opened TAN in 1952 for the Aircraft Nuclear Propulsion (ANP) Program, which operated from July 1952 until March 1961 (Stacy 2000, pp. 132, 140). Additional facilities built to support the program were the IET Facility, the Technical Support Facility (TSF), and the Water Reactor Research Test Facility (WRRTF).

During the 1970s, the Loss of Fluid Test (LOFT) Facility was built at the west end of TAN next to the aircraft hanger built to support the ANP Program. LOFT was a smaller version of a commercial pressurized-water reactor (PWR) designed to allow, create, or re-create loss-of-fluid accidents. In 1983, construction started on the classified Project X, which later became the Specific Manufacturing Capability (SMC). The SMC, which was inside the aircraft hanger, manufactured depleted uranium (DU) armor for the M1-A1 Abrams Main Battle Tank.

2.2.1 Technical Support Facility

TSF was built to provide technical and administrative support for the ANP Program. The facilities included a large Hot Shop, hot cells, storage pool, cafeteria, machine shop, office space, etc. A four-rail railroad system, with a shielded locomotive and two turntables, connected the IET Facility to the Hot Shop, the Warm Shop, and eventually the LOFT Facility.

2.2.1.1 Hot Shop, Warm Shop, and Hot Cells Annex, 1955 to Present

The TAN Hot Shop is in the TAN 607 building. It is 51 ft wide by 165 ft long by 55 ft high. The walls are 7 ft thick and the windows are 6 ft thick to provide protection to personnel during examination, handling, analysis, and disassembly of radioactive components.

The Warm Shop, immediately adjacent to the Hot Shop, is one large open room 51 ft wide by 80 ft long by 50 ft high. A four-rail track system connects the Warm Shop to the Hot Shop and supports

TAN projects. Experiments, projects, and equipment with relatively low-level contamination or direct radiation were brought to the Warm Shop for modification or repair.

The Hot Cell Annex is on the south side of the Hot Shop and consists of a hot cell and control galleries. The interior of the Hot Cell is 10 ft wide by 35 ft long. The Hot Cell Annex (TAN 633) is a one-story building north of and adjoining the storage pool. It consists of four shielded cells with interlocking sliding drawers for transferring samples.

The Hot Shop and Hot Cells are equipped with cranes, manipulators, and other equipment for remote handling and work on experiments of all types. The Hot Shop was used extensively for refueling and repairs on the Heat Transfer Reactor Experiment (HTRE) reactors. Many other activities that involved measured radiation levels up to 50 rad/hr for brief periods have occurred in the facilities, including disassembling the Stationary Low-Power Reactor No. 1 (SL-1) and receiving, examining, and storing Three Mile Island (TMI) fuel and debris [5].

Internal exposure potential exists from airborne radioactivity of mixed fission products (MFPs) and mixed activation products (MAPs) from reactor operations (Table 2-3) [6]. Small concentrations of TRU material are present (INEEL 2001, p. 125).

External exposure exists from MFPs and MAPs. Radiation levels varied from background levels to measured levels greater than 50 rad/hr, principally gamma rays with energies greater than 250 keV [7].

2.2.1.2 TAN 607 Storage Pool, 1955 to Present

The TAN Storage Pool is adjacent to the north side of the Hot Shop. It is 70 ft long, 48 ft wide, and 24 ft deep and has a capacity of approximately 500,000 gal. An underground passageway (19 ft by 24 ft) under the Hot Shop north wall connects the main pool to the Hot Shop vestibule. The vestibule, in the northeast corner of the Hot Shop, is 25 ft long, 24 ft wide, and 24 ft deep. The top of the passageway under the shield wall is 5 ft under water to protect the main pool area from radiation sources in the Hot Shop.

The TAN Pool was used for storage of reactor fuel and included some commercial fuel and TMI core debris. In mid-2002, the TMI fuel and debris were moved to INTEC for storage. Radioactive materials being put in the pool or removed are generally transferred in shielded casks. The casks can be transferred via the vestibule in the Hot Shop or loaded directly on a truck at the north end of the pool. The pool is not lined and does not meet current criteria for radioactive storage pools (Bonney et al. 1995).

The water in the Storage Pool is contaminated with MFPs, MAPs, and transuranic (TRU) materials. The major isotopes in the pool are ^{137}Cs, ^{147}Pm, ^{241}Pu, ^{241}Am, and ^{90}Sr, with an average water concentration of 4.6×10^{-4} µCi/mL (Bonney et al. 1995, pp. 9-10).

Internal exposure potential exists from airborne radioactivity, primarily from the nuclides above.

External exposure exists from the nuclides above.

2.2.1.3 Storage Pads TAN 790 and TAN 791, 1986 to Present

The Storage Pads are near the main east-west railroad west of the Hot Shop and east of the turntable. The TAN 790 pad was used to store three “Abnormal Waste Casks.” Two of the casks are

empty; the third contains filter elements with activity greater than or equal to 100 nCi/g of TRU materials (Bonney et al. 1995, p. 8). External radiation levels are less than or equal to 0.1 mrem/hr [8].

Storage Pad TAN 791 is west of TAN 607 and contains the "Spent Fuel Storage Casks." These casks were designed to investigate the feasibility of storing spent nuclear fuel from commercial reactors in a dry state without external cooling requirements. The casks have fuel from a Virginia Electric & Power Company reactor and from LOFT. The casks, which are filled with helium gas and sealed, are monitored for temperature and pressure, and alarms sound if either deviates from accepted limits (Bonney et al. 1995, p. 8). Measured radiation levels near the casks are 25 to 30 mrem/hr gamma and about 40 mrem/hr neutron. The casks are periodically surveyed and assessed for radioactive contamination [9].

Internal exposure potential exists from potential airborne leaks [10].

External exposure exists from activities near the casks. Not all personnel wear neutron dosimetry [11].

2.2.1.4 Radioactive Liquid Waste Disposal System, 1959 to Present

A radioactive liquid waste system (TAN 666), which was built in 1959, collected, processed and had interim storage capacity for all intermediate-level radioactive liquid wastes from TAN and transferred them to one of three underground 10,000-gal, stainless-steel collection tanks (V1, V2, and V3). The liquid wastes were concentrated in an evaporator in TAN 616, and the concentrated solution was pumped to one of two 50,000-gal underground liquid waste feed tanks (V9 and V10). Solids were separated and sent to the RWMC. The liquid effluent was originally combined with low-level radioactive liquid waste and discharged to a disposal well. In 1972, INL replaced the disposal well with a disposal pond. The pond was an unlined diked area encompassing about 35 acres that could receive about 33 million gal/yr (ERDA 1977). From 1959 to 1974, TAN reported liquid effluent releases to the disposal well or pond of 58 Ci, with highest releases in 1959, 1968, and 1969 (ERDA 1977, p. 143).

Internal exposure exists from potential airborne MFPs and MAPs (Bonney et al. 1995).

External exposure exists from activities around the liquid waste system [12].

2.2.1.5 Radioactive Parts Service and Storage Area, 1955 to 2004

Storage buildings (TAN 647 and TAN 648) and the adjacent storage pads are known as the Radioactive Parts Service and Storage Area (RPSSA). The facility is in the northwest portion of the TAN TSF area. The area has residual contamination from earlier projects, including the HTRE-2 and HTRE-3 reactors. The buildings and contaminated areas are marked with perimeter fencing and warning signs. Some of the radiologically-contaminated soil in the area resulted from movement of SL-1 debris into the Hot Shop, and some particles were measured as high as 50 rad/hr beta near contact (^{90}Sr/Y). The gates and buildings are locked when access is not required [13].

The RPSSA buildings and areas on and around the pads contain casks, boxes, and equipment of all types that is either contaminated or radioactive and has to be in a controlled storage area.

Internal exposure potential exists from airborne radioactivity from leaking packages or disturbed contaminated soil areas that contain MFPs and MAPs [14].

External exposure comes from several items in storage that have had radiation levels in the roentgen-per-hour range from MFPs and MAPs [15].

2.2.1.6 TAN 607 Radiography Facility

The Radiography Facility was in TAN 607, south of the TAN Warm Shop. The facility used ^{60}Co and ^{192}Ir sources and X-ray units for nondestructive examination of metal welds, parts, and equipment. Some of the isotopic sources were in the 100-Ci range when purchased [16]. Radiography was performed by trained and certified technicians in accordance with approved procedures. Periodic radiography would be required on a reactor or piece of equipment that could not be brought to the facility. Applicable procedures were required to evaluate radiological hazards and establish necessary controls. The Radiography Facility is not operational at present.

Internal exposure is negligible for radiographic activities in the Radiography Facility [17].

External exposure occurred from activities with radiographic sources [18].

2.2.2 Initial Engine Test Facility, 1955 to 1966

The IET Facility (TAN 620) was approximately 1 mi north of the TAN TSF area. It was built as a shielded test facility to prove that heat from a nuclear energy source could run a turbojet engine. The IET Facility had high-density reinforced concrete walls 2 ft thick and ceilings 3 ft thick. The floor of the facility was 15 ft below grade with 14 ft of dirt over the top. After a HTRE test, a locomotive driver in a shielded cab would hook onto the reactor and return it to the TAN TSF Hot Shop. There were a total of 26 IET runs that involved three separate reactor assemblies – HTRE 1, 2, and 3 (DOE 1991a, p. 32). Jet engines were fitted to an HTRE reactor at TSF and transported by the four-rail system to the IET Facility.

The test series involved a number of power reactor operations that resulted in the release of radionuclides to the environment. When reactor operations occurred and the consequent releases varied throughout the duration of the project due to strict meteorological control; permissible wind directions seriously limited operations. On many days, it was impossible to operate at all, and most of the time operation was possible only a few hours each day (Thornton and Rothstein 1962). Therefore, the times of the tests depended on the meteorological conditions (Stacy 2000, pp. 134-138).

During the early 1960s, the AEC initiated another nuclear safety program at the IET Facility that was called Systems for Nuclear Auxiliary Power (SNAP) Transients (SNAPTRAN). The program evaluated the hazards of using nuclear reactors for aerospace auxiliary power systems. In addition, it was designed to investigate the consequences of a nuclear accident. Three SNAPTRAN tests were conducted, and the last two ended in destruction of the SNAP 10A/2 reactors (Stacy 2000, pp. 178, 281-282).

Internal exposure was possible from airborne MFPs and MAPs. It was minimized by a filtered intake air system that kept the air pressure in occupied areas positive in relation to outside air. The exhaust from the HTRE jet engines was vented up a 150-ft stack (Thornton and Rothstein 1962, p. 154; Stacy 2000, pp. 134-138).

External exposure occurred from the MAP and MFP inventory after tests were completed when personnel worked on the associated test assemblies [19].

2.2.2.1 Heat Transfer Reactor Experiment No. 1, December 27, 1955, to January 3, 1959

On November 4, 1955, the HTRE-1 reactor was placed on the test pad at the IET and brought to power. On December 30, 1955, the reactor was again brought to power with a J47 turbojet engine attached (Stacy 2000, p. 135). The reactor was run on the test stand for 150.8 hours at full power, 20 MWt, which exceeded the design requirement of 100 hours. During the first 6 hours of full-power operation, fuel element damage occurred in three cartridges that was caused by a defect in the insulation liners. After the damaged elements were replaced, power operation resumed. The test was successful in proving the reactor could run the engines without chemical fuel. The water-moderated and water-cooled reactor used enriched uranium fuel clad in nickel-chromium (Thornton and Rothstein 1962, p. 39).

2.2.2.2 Heat Transfer Reactor Experiment No. 2, July 1957 to March 28, 1961

The HTRE-2 parent core was similar to the HTRE-1 core except that the central seven air tubes were removed and replaced by a hexagonal void 11 in. across the flat. A corresponding opening was made in the top shield plug so sections of advanced reactors could be inserted in the parent core without removal of the core from the shield. This converted it to a materials test reactor, which subjected test fuels to environments reaching 2,800°F for extended periods and higher temperatures for short periods. The ANP Program advanced the technology of high-heat ceramic reactor fuels. HTRE-2 operated for 1,299 hours at powers up to 14 MW. Test sections consisted of metallic fuel elements combined with air-cooled zirconium hydride moderators and beryllium oxide fuel elements for use in ceramic reactors (Thornton and Rothstein 1962, pp. 42-49; Stacy 2000, pp. 135, 277).

2.2.2.3 Heat Transfer Experiment No. 3, 1958 to December 1960

A new HTRE engine was developed with the reactor, engine shielding, and heat transfer system arranged in a horizontal configuration to anticipate final design in an airframe. HTRE-3 operated for 126 hours and ran two modified J47 jet engines at power as high as 32 MWt. This reactor was water-cooled, water-moderated, and used uranium fuel clad in nickel-chromium. In December 1960, HTRE-3 ran two turbojet engines at 2,000°F without the help of any chemical fuel (Thornton and Rothstein 1962, pp. 50-58; Stacy 2000, pp. 137, 278).

On November 18, 1958, a nuclear excursion occurred during IET No. 13. The excursion occurred when the dynamic and shim rods were withdrawn by the control system under the influence of an erroneous reactor power indication. Activity was released from the exhaust stack and a narrow band of fallout was contained fully within the INL boundaries. The maximum dose rate that was observed in the Assembly and Maintenance area and approximately 3,000 ft from the cloud centerline was 0.04 mrem/hr. The maximum fallout observed, at about 4 hours after the incident, measured 0.8 to 2.0 mrem/hr at contact roughly 1.5 mi from the IET Facility (DOE 1991a, pp. 143-144).

After each HTRE test, the cooling water was drained and replaced with mercury to provide shielding to personnel who worked on the reactors (Stacy 2000, p.137). After the HTRE program ended, HTRE-2 and -3 were parked in the RPSSA. The HTRE engines were moved to the EBR-I historic site in 1988.

Exposures from all experiments:

Internal exposure potential existed from the airborne release inventory during reactor operation as argon and other constituents in the cooling air became radioactive. Fuel elements occasionally ruptured, which released MFPs and MAPs to the outdoor environment. In some cases the cooling

airflow to fuel elements was deliberately blocked to determine fuel failure parameters and characteristics. Fourteen of the tests were categorized as Group 1 releases under National Council on Radiation Protection and Measurements methodology (NCRP 1996); tests 10 and 4 were ranked the highest. Seven tests were ranked as Group 2. Fission product inventories were based on documented reactor operating histories (Till et al. 2002). Operations and support personnel were inside the pressurized control room (TAN 620) during reactor operation during the testing phase of the HTRE program, which minimized their exposure potentials.

External exposure occurred from the MFP and MAP inventory during work with test assemblies after tests were completed [20].

2.2.2.4 Systems for Nuclear Auxiliary Power 10A Transient No. 1, Early 1960s

The AEC initiated a safety program to evaluate the hazards of using nuclear reactors for aerospace power systems. The portion of the program that was concerned with determining the kinetic behavior of the SNAP 10A/2 reactors and the consequences of certain nuclear accidents that involved them was called SNAPTRAN. The tests were conducted at the IET Facility at TAN. The SNAPTRAN program extended the SPERT reactor safety testing program (Section 2.9) to aerospace applications. Three test series that involved three reactors investigated the behavior of SNAP 10A/2 fuel under large-transient, power-excursion conditions. SNAPTRAN-1 was subjected to nondestructive tests in conditions that approached but did not result in damage to the zirconium-hydride-uranium fuel (Stacy 2000, pp. 281-282). The SNAP 10A/2 reactors were 9 in. in diameter by 12 in. long and were composed of a sodium-potassium (NaK)-cooled core with 37 rods of fully enriched uranium in a zirconium-hydride matrix (Cordes et al. 1967, p. 8).

2.2.2.5 Systems for Nuclear Auxiliary Power 10A Transient No. 3, April 1, 1964

SNAPTRAN-3 was the first of two destructive tests on a version of the small SNAP 10A/2 reactor, which was designed to supply auxiliary power in space. The test was conducted at the IET Facility on April 1, 1964, and simulated the accidental fall of a reactor into water or wet earth, as could occur during assembly, transport, or launch abort. The test demonstrated that the reactor would destroy itself immediately rather than build up a high inventory of radioactive fission products (Stacy 2000, p. 282). The test involved reactor operation at a power level of 30,000 MW for 1.5 milliseconds. More than 99% of the fission product inventory was retained in the surrounding water and reactor fuel remains. No airborne iodine was detected, so it was presumed that halogens were retained in the water as well as particulate radionuclides (Cordes et al. 1965, pp. 60-61).

2.2.2.6 Systems for Nuclear Auxiliary Power 10A Transient No. 2, 1965 to January 11, 1966

This test version of the small SNAP 10A/2 space reactor was intentionally destroyed on January 11, 1966. It provided information on the dynamic response, fuel behavior, and inherent shutdown mechanisms of these reactors in an open-air environment. In normal operation, the control drums of the SNAP 10A/2 would be rotated to obtain criticality after the reactor was in orbit. In a launch abort, however, impact on the earth might cause the reactor drums to rotate inward and the core to go critical and conceivably destroy itself, which would release fission products to the surrounding environment. The test data contributed to an understanding of reactor disassembly on impact and methods for assessing or predicting the radiological consequences (Stacy 2000, p. 282). The reactor core was 93%-enriched fuel and contained 4.75 kg of ^{235}U. The reactor operated at a peak power of 74,000 MW and a total of 54 MW-s of nuclear energy was released for 1.5 milliseconds before and after the peak power. The fission product release fractions were reported to be 0.75 for noble gases, 0.70 for iodines, 0.45 for tellurium, and 0.04 for solids. Reactor operation was assumed to generate

4,681 Ci of ^{41}Ar. SNAPTRAN meteorological conditions for testing were strict. Weather requirements were to consist of lapse conditions with no rainfall and were to persist a minimum of three (3) hours after the tests. Wind direction had to be from the southwest (180˚ to 240˚) between 10 and 30 miles per hour (Cordes et al. 1967).

Exposures from all tests:

Internal exposure potential during the SNAP 10A test series was well controlled by requiring personnel to stay inside the IET Facility during tests. During the SNAPTRAN-2 and -3 tests, the reactors were destroyed and released fission products including ^{131}I, all of which was tracked (Cordes et al. 1965, 1967).

External exposure to MFPs and MAPs was received during cleanup of the reactor debris, which was scattered around the test pad and along the four-track rail system. Dosimetry was required for all personnel [21].

2.2.3 Water Reactor Research Test Facility, 1958 to 1973

The WRRTF is approximately 1.25 mi southeast of the TSF area. The first facility at WRRTF was the Low Power Test Facility (LPTF) in 1958. This facility, also known as Semiscale, was in the east quadrant of WRRTF. The Shield Test Pool Facility (STPF) in the west quadrant of WRRTF was built as part of the ANP Shielding Experimentation Program; in 1963, the pool facility was modified for the Experimental Beryllium Oxide Reactor (EBOR). The EBOR project was cancelled in 1966 before construction was complete (Stacy 2000, p.276).

The Semiscale facility in LPTF was a forerunner of the LOFT. It was a scaled mockup of one loop of a four-loop PWR. The facility was electrically heated to provide steam to run blowdown tests (Shaw, Boucher, and Loomis 1985). No radiological exposure resulted from the blowdown tests [22].

Due to the electrically heated experiment cores in the facilities that are discussed in the following sections, internal and external exposure from neutron reactions in the reactors did not exist [23]. Applicable exposure potentials are noted at the end of the pertinent test series.

2.2.3.1 Shield Test Pool Facility

The STPF was initially used for a reactor experiment known as SUSIE and then for bulk shielding experiments in support of the ANP Shielding Experimentation Program. SUSIE utilized a sample canister box that was pressurized with inert gas or air to keep it dry. It was at the reactor centerline and contained 16 sample tubes in which organic samples were irradiated. SUSIE was a swimming-pool-type reactor; that is, it was water-moderated, water-cooled, water-reflected, and shielded by approximately 17 ft of water. The fuel loading was 4 kg of ^{235}U, and the nominal power level was 2 MWt. After the ANP Program ended in 1961, SUSIE was used by other programs at INL (Walsh 1961) and would later become known as EBOR. After the EBOR project was cancelled, other experiments were conducted using tracer-level nuclides to investigate the reaction of water in piping, and instrument calibrations were performed. The cell area was filled with piping for pressurized-water tests using up to 25-Ci ^{137}Cs sources in a radiography-type environment in which the source material was cranked out of a shielding cask to be in proximity to the piping. The cesium and tracer sources have been removed [24].

Internal exposure at the STPF was not known to occur due to the low power of SUSIE, the water environment, and use of sealed small sources [25].

External exposure occurred from the use of sealed sources, primarily ^{137}Cs [26].

2.2.3.2 Low Power Test Facility, 1958 to 1973

The LPTF was used to conduct several low-power (less than 100-W) reactor research programs. The LPTF contained two shielded cells with three independent control rooms and necessary support facilities. The north cell, Room 101, was called the Critical Experiment (CE) cell and the south cell, Room 102, was the Initial Criticality cell. The test cells are of poured concrete construction with a 4-ft-thick wall between them. The walls between the cells and the control room are 5 ft thick and 30 ft high. The outside wall of the Initial Criticality cell is 2 ft thick, and the outside wall of the CE cell is 3 ft thick and 30 ft high (Kunze and Chase 1970).

The construction of the facility was such that more than one reactor program could be running at the same time [27]. Heavy experiment pieces could be moved in or out of each cell through large rollup doors in the back.

The following exposure potentials are for all subsections of Section 2.2.3.2.

Internal exposure was not known to occur [28].

External exposures occurred during cell entry after a reactor run [29].

2.2.3.2.1 Critical Experiment Tank, 1958 to 1960

The Critical Experiment Tank (CET) reactor was part of the ANP Program in the CE Cell of the LPTF. The CET was a low-power reactor that was originally designed to mock-up the HTRE-1 and HTRE-2 reactors. It was used primarily to perform critical experiments for insert tests in the HTRE-2 power plant. Fuel test bundles for testing in HTRE-2 were first evaluated for reactivity characteristics in the CET. The water-moderated CET used a beryllium reflector (Becar et al. 1961).

CET was one of three low-power reactors that supported the ANP Program along with the STPF Reactor (SUSIE) and the Hot Critical Experiment (HOTCE) (Hoefer 1957). The ANP Program ended in March, 1961 (Stacy 2000, p. 140).

2.2.3.2.2 Hot Critical Experiment, 1958 to March 28, 1961

In the LPTF CE cell, HOTCE was an elevated-temperature critical experiment that was designed to obtain information on temperature coefficients of solid moderated reactors. The fuel elements consisted of fuel bearing stainless-steel wire 1/8 in. in diameter. The maximum loading was 50.4 kg of 93.2%-enriched UO_2. The reactor used a hydrided zirconium moderator and a beryllium reflector. The hexagonal prism-shaped core and reflector were mounted such that the fuel cells were horizontal. One half was mounted on a fixed table and the other in a movable table so the two halves were separate. The normal operating power was 1 W for a period of 1 to 3 hours. The reactor could be operated at 100 W for short periods (Hoefer 1957).

2.2.3.2.3 Split Table Reactor System, 1971

The purpose of the Split Table Reactor System was to provide information on a variety of fast and thermal spectrum reactors. The reactors were to be assembled, operated, and revised to perform experiments with both thermal and fast systems. The reactor operated in the CE cell of the LPTF.

The reactor was a split-table type, 7 ft 2 in. wide and 11 ft long. A hydraulic system opened and closed the reactor. The table was an aluminum matrix structure that was composed of a stack of hexagonal tubes mounted horizontally on each table half. When the two halves were brought together, a single reactor assembly was formed. Normal operating power level was 0 to 500 Wt not to exceed 1 kW or 10 kW-hr per month (Lofthouse 1971).

2.2.3.2.4 Fast Spectrum Refractory Metals Reactor (710), March 1962 to 1968

The 710 reactor was a split-table, low-power critical facility at LPTF. The objective was to collect data for a proposed fast-spectrum, refractory-metal reactor concept called the 710 Reactor. The concept involved using metals such as tungsten and tantalum in a compact, very-high-temperature reactor for generating power in space (Stacy 2000, p.277).

2.2.3.2.5 Cavity Reactor Critical Experiment, May 17, 1967, to Early 1970s

The Cavity Reactor Critical Experiment (CRCE) was an outgrowth of a program that was started by the National Aeronautics and Space Administration at LPTF to investigate the propulsion of space rockets by nuclear power; it offered the possibility of much greater thrust per pound of propellant than chemical rockets. The concept for the cavity reactor core was that the uranium would be in a vapor or gaseous state. Hydrogen propellant flowing around it would theoretically attain much higher temperatures (up to 10,000°F) than in conventional solid-core rockets. The experiments at TAN used simulated hydrogen propellant and produced data on the reactor physics feasibility of a gaseous core able to go critical. The core was uranium hexafluoride (UF_6); the experiments were performed at the relatively low temperature of about 200°F. In the proposed ultimate application, the ball of uranium gas would have been held in place by the hydrogen flowing around it, something like a ping-pong ball suspended in a stream of air. Uranium core temperatures as high as 100,000°F were considered possible. During the CRCE program, the UF_6 was always contained in the storage vessel or the reactor cavity (Stacy 2000, p.275).

2.2.3.2.6 Spherical Cavity Reactor Critical Facility, 1972 to November 1973

The Spherical Cavity Reactor Critical Facility was the final experiment in reactor physics work for the National Aeronautics and Space Administration program to determine the feasibility of a reactor going critical with a gaseous core of UF_6. Previous work used a cylindrical configuration because of its ease of construction. The spherical shape was considered a more likely geometry for the ultimate application in a rocket to Mars (Stacy, p. 281). The assembly consisted of two aluminum tanks, one inside the other, with D_2O in the space between the two tanks. The D_2O acted as a reflector and moderator and, during normal shutdown, would be transferred from the reactor to a storage tank (INC 1969).

2.2.3.2.7 High Temperature Marine Propulsion Reactor (630-A), 1962 to 1964

The 630-A reactor was a low-power critical experiment at the LPTF. The mission of the 630-A was to explore the feasibility of an air-cooled, water-moderated system for nuclear-powered merchant ships. Development ended in December 1964 after decisions to lower the priority of the entire Nuclear Power Merchant Ship Program (Stacy 2000, p.278).

2.2.4 Loss-of-Fluid Test Facility, 1973 to July 9, 1985

The LOFT reactor at TAN 650 was a centerpiece in the safety testing program for commercial power reactors. The reactor was a scale model of a commercial PWR that was built to explore the effects of

loss-of-coolant accidents (LOCAs). Thirty-eight nuclear power tests were conducted with various accident scenarios, including the accident at TMI. Among other goals, the program investigated the capability of emergency core cooling systems to prevent core damage during a LOCA. Experiments at LOFT simulated small-, medium-, and large-break LOCAs, sometimes complicated with other events such as loss of offsite power. LOFT was deactivated in 1986 after completion of the LP-FP-2 experiment, which was the most significant severe fuel damage test ever conducted in a nuclear reactor. That test, which involved the heating and melting of a 100-rod experimental fuel bundle, provided information on the release and transport of fission products that could happen during an actual commercial reactor accident where core damage occurred (Stacy 2000, p.278).

The LOFT facility is in a steel-domed reactor containment vessel 97 ft high with a basement. The LOFT Control and Support Building is four stories high with a basement. The structures are attached at the basement level, and both have reinforced concrete exterior walls.

The control room, visitor center, experimental data recording and display area, sample counting area, and operation support room were in TAN 630, a two-story underground building. During the final preparations for a test, the containment vessel doors were closed and the only access to the facility was through a shielded underground tunnel. During reactor operation and testing, personnel were restricted to safe facilities in TAN 630.

Internal exposure was possible from airborne fission product activity in the containment soon after shutdown. Entries were monitored with a continuous air monitor, and respiratory protection was worn as required. On July 9, 1985, after completion of the LP-FP-2 test, leakage was discovered from the fission product monitoring system and the primary coolant system, which allowed fission products to enter the reactor building. Over the following 2-month period, 8,780 Ci of noble gas (^{88}Kr) and 0.09 Ci of iodine (^{131}I) were released to the environment (Hoff, Chew, and Rope 1986, Table B-10, p. 53).

External exposure occurred to personnel who worked inside the containment vessel or on the reactor's primary system or the sample systems. During initial entry after a test, the fields in containment were ≥100 mrad/hr beta-gamma. The short-lived fission products would decay rapidly and reduce the general fields to ≤10 mrad/hr beta-gamma [30].

2.2.5 Specific Manufacturing Capability, 1985 to Present

The SMC project is in the ANP Program aircraft hanger (TAN 629) and surrounding buildings. The project consists of Phase I, Phase II, and support facilities. A Materials Development Facility in TAN 607A and the TAN Hazardous Waste Storage Area in TAN 628, were part of the SMC project, but have been decommissioned and turned over for other use. The SMC project is classified. In 1991, the mission was declassified in that the SMC manufactures armor made of DU for the U.S. Army. The major radioactivity in the DU is ^{238}U, ^{234}Th, ^{234m}Pa, and ^{234}U (INEEL 2001, p. 54).

The SMC facilities consist of Phase I in TAN 629 and Phase II in TAN 679 and TAN 681. Metal fabrication activities are performed in Phase I facilities. Phase II facilities perform metal rolling in TAN 679 and waste processing in TAN 681. SMC formerly used a nitric acid system; this waste was processed in TAN 681. This system was removed and replaced with an aqueous system. The aqueous waste is processed in TAN 681. All radioactive aqueous waste is collected in storage tanks for treatment through an evaporator system, and the remaining aqueous waste is solidified and disposed of as low-level radioactive waste.

Internal exposure occurred from normal operations using the DU processes. Metal fabrication is the primary source of airborne radiological activity, followed by a paint coating process [31].

External exposure occurred through working with the billets of DU. The large pieces were primarily handled remotely to minimize exposure [32].

2.3 IDAHO NUCLEAR TECHNOLOGY AND ENGINEERING CENTER, NOVEMBER 1951 TO PRESENT

INTEC, formerly the ICPP and commonly known as the Chem Plant, is 53 mi west of Idaho Falls and occupies 200 acres in the middle of the INL reservation. INTEC stored and processed spent nuclear fuel from university and test reactors all over the world, from commercial power plants, from most of the reactors at INL, and from U.S. Department of Defense projects. INTEC received the first fuel shipment in November 1951. The first hot run started in February 1953 and reprocessing continued until the 42nd run was completed in July 1988. The fuel reprocessing project was cancelled in 1992 and a clean-out campaign was completed in 1996 (Wagner 1999, p. 17). Stacy (2000), Lewis et al. (2000), and Wagner (1999) summarized the significant events during the operation of the Fuel Reprocessing Center.

The primary INTEC mission involved reprocessing spent nuclear fuel with HEU, which entailed extracting reusable uranium from spent fuels. Each fuel type or cladding (e.g., aluminum, zirconium, stainless steel, and graphite) called for a different process. The numerous fuel recycling processes required support facilities for fuel dissolution and recovery of fissionable materials (e.g., there were 29 different process cells) and waste processing. An innovative high-level liquid waste (HLLW) treatment process known as calcining was developed at INTEC as an additional major mission. Calcination reduced the volume of liquid radioactive waste from reprocessing and placed it in a more stable solid granular form.

In addition to calcination and fuel reprocessing operations, INTEC is a major fuel storage facility. INTEC houses an HLLW evaporation facility, HLLW storage (Tank Farm), airborne radioactive waste processing, nonradioactive liquid waste disposal, and the Remote Analytical Laboratory (RAL).

Although fuel reprocessing ended in 1992, INTEC continues to support other nuclear projects. For example, chemical research continues to improve fuel recovery processes, spent nuclear fuel is safely stored and prepared for shipment to an offsite repository, development of technology to treat safely high-level and liquid radioactive waste from reprocessing spent fuel continues, past environmental releases are being remediated, and some facilities have undergone decontamination and decommissioning (D&D). Inactive INTEC facilities are being evaluated for D&D.

Internal exposure could have occurred when workers were near breached or leaking systems that contained MFPs, MAPs, TRU materials, etc. [33].

External exposure occurred during maintenance work, laboratory work, fuel cutting, and other support work that briefly exposed workers to radiation rates from background levels to measured levels of ≥50 rad/hr beta-gamma [34].

2.3.1 Fuel Processing Facility, CPP 601 and 602, February 1953 to 1992

The INTEC Fuel Processing Facility (FPF), CPP 601/602, was used for the chemical separation of HEU from dissolved spent fuel during reprocessing and to solidify the recovered HEU for shipment off the site. The process dissolved the fuel in acid and produced uranyl nitrate, nitrates of fission products, and some TRU materials. Solvent extraction with hexone and tributyl phosphate (Boardman ca. 1956) separated uranium from the fission products.

The Process Building, CPP 601, contains 29 heavily shielded underground process cells. The building is 250 ft long by 102 ft wide, extends to 57 ft below grade and 38 ft above grade, and features at least seven corridors for different functions (Cederberg et al. 1974, Wagner 1999, p. 17, Knecht et al. 1997, p. 3). Table 2-4 lists the cells and processes for fuel reprocessing. With the exception of the Health Physics field office in V cell, cells were accessible only during shutdown periods for maintenance and decontamination activities. The process building was designed to handle modest quantities (up to several kilograms) of enriched spent fuel due to criticality considerations. It was designed to be remotely decontaminated so hands-on maintenance could occur. The use of hands-on maintenance resulted in many activities being conducted with significant dose rates and contamination levels [35].

The Laboratory Building, CPP 602, shares a common wall with CPP 601 and is 147 ft long, 102 ft wide, and about 80 ft high (much underground). It is used to support activities in the process building.

The final product was uranyl nitrate solution essentially free of impurities and fission products. It was shipped to the Y-12 Plant in 10-L polyethylene bottles in concentrations above 250 g U/L (Lewis et al. 2000). Beginning in 1971, a denitrator in a glovebox in CPP 602 was used to convert the uranyl nitrate to solid UO_3 using a fluidized bed thermal conversion process.

Table 2-4. INTEC 601 and 602 process cell information (Cederberg et al. 1974, Table 3.1-I, pp. 23-25; Stacy and Braun 2006, pp. 65-73).

Cell	Process description	Cell function
A	EBR feed preparation	Fuel dissolution
C	MTR feed preparation	Fuel dissolution
D	MTR feed preparation	Fuel dissolution
E	Submarine thermal reactor (S1W) feed preparation	Fuel dissolution
F	First-cycle extraction	Uranium separation
G	MTR feed preparation	Fuel dissolution
H	MTR first-cycle extraction	Uranium separation
J	Hot salvage	
K	Solvent recovery	
L	RaLa process cell	Recovery of RaLa
P	Second-cycle extraction	
Q	Third-cycle extraction	
R	Product transfer cell	
S	Fourth-cycle extraction	
T	Solvent pumps	Hexone storage
U	Second-cycle aqueous raffinate treatment	
V	Health Physics field office	
W	Second-cycle solvent raffinate	
X	Sample dilution and decontamination	
Y	Third- and fourth-cycle raffinate treatment	
Z	Product room	

In 1956, a process for the recovery of radioactive ^{140}La (RaLa) was assigned to INTEC. The RaLa process took place in the complicated L Cell and lasted into 1963. As freshly irradiated Materials Test Reactor (MTR) fuel was dissolved in acid, the dissolving process liberated gases, one of which was ^{131}I, with an 8-day half-life. The RaLa process recovered ^{140}Ba for its ^{140}La progeny and shipped it to Los Alamos National Laboratory for use in weapon research projects [36] (SCA, SRA and WCC 1994, p. 2-9).

Beginning in 1965, neptunium was collected from the second-cycle partitioning step. In 1972, this material was cleaned up using two cycles of hexone. Approximately 5.4 kg of neptunium was shipped to the Savannah River Site for use as targets in making ^{238}Pu in this demonstration project. The processing was done in CPP 601, and bottling of the product was done in the multicurie cell in CPP 627. The process did not separate the neptunium, so it remained a minor constituent of the product and waste [37] (Stacy and Braun 2006, pp. 39-40).

When reprocessing was discontinued in 1992, these facilities were flushed to remove uranium and hazardous materials and placed in a standby condition. The Fuel Processing Restoration (FPR) Project would have replaced these facilities. The project was about 40% complete when construction stopped in 1992 and discontinued in a manner that preserved the facility for possible use in future missions at INTEC.

Three criticality events occurred during FPF operation:

- Criticality Accident of October 16, 1959 (Ginkel et al. 1960). A bank of storage cylinders that contained dissolved spent EBR-1 fuel elements with enriched uranium was air-sparged (air was bubbled violently into the solution to mix it). The cylinders were geometrically safe, but the sparging initiated a siphon that transferred 200 L of the solution to a 5,000-gal tank that contained about 600 to 900 L of water. The resultant criticality lasted about 20 minutes. No workers were exposed to gamma or neutron radiation from the criticality because the criticality occurred in an unoccupied belowground cell. Airborne activity with some entrained liquid spread through the plant through vent lines and drain connections, which triggered alarms and an evacuation. Pressure in the vessel with the criticality removed about 900 L of solution (76 L remained in the vessel) and unexplainably moved about 600 L into a companion vessel. A high radiation level (>25 R/hr) was discovered near the RaLa area above the waste tank (Ginkel et al. 1960). Fields beyond the guardhouse measured 2 R/hr. The high fields were probably due to the large activity in the dissolved spent fuel outside its intended location. Two people who evacuated received internal exposures (<30 mrem) as they passed the evacuation route area where radioactive gas was being released into the room from floor drains. Twelve evacuees received film badge doses from 2 to 50 rem (beta and gamma, mostly beta) (Ginkel et al. 1960 p. 44).

- Criticality Accident of January 25, 1961. About 40 L of uranyl nitrate solution (200 g U/L) were forced upward from a 5-in.-diameter section of an evaporator into a 24-in.-diameter vapor disengagement cylinder, well above normal solution level. Analysts later assumed that air entered associated lines while operators were attempting to clear a plugged line and improve pump function. When the air bubble reached the evaporator, solution was expelled from the lower section and a momentary criticality occurred in the upper section. The radiation triggered alarms, but no personnel received ≥100 mrem exposure (Stacy 2000). Concrete shielding walls surrounded the location of the criticality, the vent system prevented airborne activity from entering work areas, and equipment design prevented a persistent excursion (Paulus et al. 1961).

- Criticality Accident of October 17, 1978. A criticality event occurred in the first-cycle tributylphosphate extraction system in the CPP 601 process building. The incident resulted in no personnel injury, no onsite or offsite contamination, and no damage to equipment or property. The plume traveled over uninhabited areas to the southwest of the site (Casto 1980). The criticality caused approximately 3×10^{18} fissions of ^{235}U. The Atmospheric Protection System at INTEC, which became operational in 1975, significantly reduced particulate emissions and filtered all releases from the criticality event (ERDA 1977, p. II-67).

Internal exposure potential existed at the FPF and its support facilities from work with radioactive airborne particulates during maintenance activities, piping and valve changes, the criticality accidents, and stack releases [38].

External exposure occurred from work in Radiation Areas and High Radiation Areas that contained the nuclides in Table 2-3. Brief exposures to measured levels of 50 rad/hr existed in the cells during piping and valve changes, decontamination, maintenance activities, working of production samples in the laboratories, etc. [39].

2.3.2 Ancillary Facilities

New fuels presented special problems in reprocessing. The Process Improvement Facility (CPP 620 and CPP 637) and the Hot Pilot Plant (HPP, later Headend Processing Plant; CPP 640) were used to provide information to improve these processes.

The Process Improvement Facility includes a laboratory building with office space for the technical group. New ideas in reprocessing are developed and investigated at the laboratory bench scale. The laboratories were designed to handle as much as 1 Ci of the radioactive materials from Table 2-3 per laboratory. The waste stream discharged to the HPP waste system.

The HPP was used to test unproven equipment and systems. The facility consisted of five cells and associated utilities. The partition between two of the cells could be removed to make one large cell. The shielding around the cells was equivalent to that of the main plant and provided sufficient radiation shielding to run plant-level radioactive material. The graphite-based ROVER nuclear rocket fuels were processed beginning in 1983 for 14 months in CPP 640 using two stages of burning to reduce the carbon content (Knecht et al. 1997). The ash was leached with a nitric-hydrofluoric acid mixture, extracted through three cycles of extraction, and converted to UO_3. More than 100 kg of uranium were still in the ash at CPP after the burners were cleaned out in 1998.

The HPP waste system consists of three-level storage. HLLW in storage can be routed to permanent storage tanks. Intermediate-level waste can be routed to either the waste evaporator system or the low-level waste tanks. Low-level waste is monitored and discharged to the disposal well downstream from the main service waste monitoring system. A service waste monitoring system is provided for this secondary stream.

The Remote Analytical Facility in CPP 627 houses the RAL, the multicurie cell, a radiochemistry laboratory, and a decontamination facility.

The decontamination facility in CPP 627 provided support for cleaning tools and equipment for INTEC and other INL facilities. Items such as water pit gates, cooling pumps, vehicles, etc. were decontaminated in this facility. It was also used as a morgue and autopsy facility following the SL-1 accident. Radiation levels to 25 rem/hr open-window beta-gamma were experienced for brief periods in the decontamination facility. Shielding other than temporary was not provided between the workstations [40].

The RAL provides two rows of 15 analytical boxes behind a 9-in. iron shield wall (Stevenson and Lyon 1955). The boxes can be remotely replaced to provide changed analytical capability. A sample transfer system below the boxes provides remote handling of samples brought to the boxes. The equipment is operated by hand-operated manipulators that extend through the shielding, pneumatically, or with electronic controls. This facility has been used for remote examination of hazardous and radioactive materials to support INTEC operations. It continues to support INL and

INTEC activities. The RAL handles, processes, analyzes, and experiments on hazardous and radioactive materials of all types. Samples that were collected in stations in the CPP 601 sample corridor were transferred to the RAL for analysis. Frequent 10-mL samples during fuel reprocessing had radiation readings of greater than or equal to 25 rad/hr. During the analytical process the samples were diluted to reduce personnel exposure [41].

The multicurie cell, which has walls 5 ft thick of barytes concrete, is designed to reduce the field from 75,000 Ci of 1.6-MeV gamma emitter to 1 mR/hr (Boardman ca. 1956). There is also a radiochemistry laboratory nearby to support operations in the multicurie cell. A walk-in hood in this area contained the custom processing facility. On February 9, 1991, a small explosion destroyed the 6-in.-outer-diameter borosilicate glass dissolver section and contaminated four employees and a portion of the laboratory. Internal exposures ranged from 0.24 to 9.1 mrem/yr for 50 years. The unirradiated material came from a cleanup campaign at ANL-East and is suspected to have contained zirconium, which would react explosively with the nitric acid that was used to dissolve the uranium (Decker 1991).

2.3.3 INTEC Fuel Storage Facility (CPP 603), 1950 to Present

The original fuel storage facility at INTEC, CPP 603, included a special fuel storage building with three 20-ft-deep storage pools for spent nuclear fuel. The facility is about one-third of a mile south of the main processing building.

Levels of airborne radioactivity of MFP around the CPP 603 unlined storage pools were a chronic problem from sodium-contaminated EBR-I fuel, which also led to contamination in the building. Efforts to clean up the water were aggravated by deionized water that corroded the concrete pool. Anticontamination apparel was provided, but respiratory protection was generally not required. Air activity was routinely measured at 10% to 25% (Rich et al. 1974) of the radioactivity concentration guideline for soluble ^{90}Sr.

In addition, CPP 603 contains the Irradiated Fuel Storage Facility (IFSF), a graphite fuel storage area, and a Fuel Cutting Facility. The IFSF stores dry fuel that is incompatible with underwater storage. The IFSF has 636 storage positions and is more than half-full. Most of the spent fuel there came from the Fort St. Vrain commercial reactor in Colorado. Shipments from Fort St. Vrain ended in 1991.

In the 1950s, INTEC received a request to process spent fuel elements from Savannah River Site reactors. The 14-ft elements were clad in aluminum and they had to be cut to 18-in. lengths to fit in the dissolver vessel. Irradiation of the fuel changed its metallurgical characteristics, so rather than cutting like regular aluminum they crumbled, which necessitated development of a new technique and procedure. Equipment change and maintenance was extremely difficult because of the crumbling, which contributed to increased levels of exposure and contamination. As a result, complete modification of the process and equipment was required.

The pools in CPP 603 were built in 1950 and served as the primary spent fuel storage facility until 1984. Fuel that was once in underwater storage at CPP 603 has been transferred to the newer underwater storage pools at the Fluorinel Dissolution Process and Fuel Storage (FAST) Facility (CPP-666) that began operation in 1984 (DOE 2007, p. 9) or to dry storage. As part of this, the CPP 603 storage pools were cleaned of sludge, dewatered, and filled with grout in 2006. A closure plan was developed to address clean closure of the FDP under the Hazardous Waste Management Act and the Resource Conservation and Recovery Act (RCRA) (DOE 2007, p. 10).

Internal exposure potential was greatest from fuel cutting and shearing, which created airborne radioactivity from the cut Savannah River Site fuel elements. A chronic exposure condition existed to MFPs from the pool water (Rich et al. 1974).

External exposure was received during fuel loading and unloading, movements to various locations in the storage pools, fuel cutting, cleanup activities, modifications, and D&D evaluations [42].

2.3.4 High Level Liquid Waste Underground Storage Tanks (Tank Farm), 1951 to Present

The HLLW Tank Farm includes 11 underground, stainless-steel, 300,000-gal storage tanks nested in concrete vaults east of process building CPP 601. The tanks were used to store radioactive liquid waste from the reprocessing of spent fuel and plant decontamination work. One tank was always kept empty for use as a transfer backup if a problem developed with one of the other tanks. All of the Tank Farm liquid has been calcined, which reduced the volume and converted it to a more stable solid form. The underground tanks are encased in concrete vaults that have sumps and leak detection. The tanks are extremely corrosion-resistant. No leakage has been detected from the tanks, but some leaks have occurred from transfer lines outside the tanks.

High-level waste at INL consists of acidic liquid and calcined solids. The acidic liquids have been stored in the underground tanks and included actual high-level waste as well as sodium-bearing waste that is managed as high-level waste. The stainless-steel tanks allowed the storage of waste in acidic form and resisted corrosion. When full, each tank contained only a few gallons of pure radioactive fission products. The rest of the solution was dissolved cladding-metal ions, process additives, and water. The tanks that received waste from the first-cycle extraction, which accumulated most of the fission products, had cooling systems to carry away decay heat to minimize corrosion (Knecht et al. 1997, Stacy and Braun 2006, p. 51).

In the 1990s, a major effort of cleanup and repair in the valve boxes of the Tank Farm resulted in a large collective dose. This project, like many others, was preplanned and reviewed by the Westinghouse Idaho Nuclear Company As-Low-As-Reasonably-Achievable Committee. The workers wore thermoluminescent and electronic dosimeters and/or self-reading pencil dosimeters. Although this dose was separately tracked, it is included in the cumulative dose records for those employees [43].

The HLLW Processing Facility, CPP 604, is east of CPP 601. Liquid waste from reprocessing activities was transferred to the Liquid Waste Evaporator, where the liquid was heated, reduced in volume, and stored in an underground tank. Soil, airborne, and groundwater contamination have resulted from these operations.

Part of the processing included a Rare Gas Processing Facility (CPP 604). Its purpose was to recover ^{85}Kr from spent fuels. In 1958, the process was enhanced by replacing the liquid nitrogen-cooled carbon beds with a cryogenic distribution system. This gas product was shipped to Oak Ridge National Laboratory for commercial sale for use primarily in leak detection. INTEC was the only source of ^{85}Kr outside the former Soviet Union (SCA, SRA, and WCC 1994).

Internal exposure potential existed from airborne radioactivity that was created during flushing operations, valve repairs, or other maintenance activities in contamination areas from MFPs and MAPs [44].

External exposure occurred during work in the contamination areas and on valve changes and piping maintenance where measured exposure rates were as high as 500 rad/hr [45].

2.3.5 Waste Calcining Facility 1 and New Waste Calcining Facility, December 1963 to 2002

To remove liquid from the waste, the AEC developed a fluidized-bed calcination process and built it at INTEC. Scientists at ANL-W tested the method in small-scale models in 1955. The process not only solidified the waste, but the product was granular, free flowing, and easily handled by pneumatic transport techniques. Phillips engineers started designing the Waste Calcining Facility (WCF) in 1956, and construction of WCF-1 started in 1958. The facility was constructed just east of the main INTEC process building and south of the storage tanks. Thick concrete shielding walls surrounded the process cells, which were below grade. The first campaign lasted until October 1964. Liquid waste was injected into a fluidized-bed chamber that was heated to 420˚C by a NaK heat exchanger system. Liquids evaporated and solids collected on the bed material, which was then collected in storage bins. Two 300,000-gal tanks and part of a third were emptied before the campaign was forced to stop because it had filled all available calcine bins. Half a million gallons of liquid had been transformed into 7,500 bulk cubic ft of solid waste. This was a reduction in volume of more than 9 to 1 (Knecht et al. 1997, p. 10). The gases leaving the stack included ^{90}Sr and ^{106}Ru, but the levels were below guideline limits (AEC 1969). In 1970, an in-bed oxygen atomized kerosene combustion system was installed to raise the bed temperature to 500˚C and reduced wall temperatures and ruthenium concentrations in the off-gas (Stacy 2000, pp. 183-186).

In 1982, the New Waste Calcining Facility replaced WCF-1. It converted HLLW from the Tank Farm into a granular solid similar in consistency to sand. The HLLW was drawn from underground storage tanks and sprayed into a calciner vessel that was superheated by a mixture of kerosene and oxygen. The liquid evaporated while radioactive solids adhered to the granular bed material in the vessel. The off-gases were treated and monitored before release to the environment, and the residual calcine solids were transferred to large stainless-steel structures that were encased in thick concrete vaults called bin sets (Knecht et al. 1997). The calciner was shut down in May 2000 while DOE evaluated whether to upgrade it to meet new emissions standards or to develop a new technology to treat the remaining liquid in the Tank Farm. The calciner operated one last run in 2002 to eliminate the remaining HLLW in the 300,000-gal storage tanks. HLLW typically contained 300 Ci/m^3 [46] (DOE 1997, p. 47).

To date, all HLLW has been removed from the Tank Farm and solidified through calcination. Removing the sodium-bearing waste remains one of the highest DOE priorities.

The New Waste Calcining Facility was the location of a decontamination facility for cleanup of radiologically-contaminated materials from INTEC and occasionally from other INL processes [47]. The decontamination facility continues to function to support INL cleanup activities.

Internal exposure potential existed from releases of contamination to the occupational environment due to leaks from piping breaks, equipment failures, and other actions that permitted unplanned releases and from decontamination activities that created airborne radioactivity [48].

External exposure resulted from routine maintenance on the transfer piping and associated valves and equipment. Calcine process cell entries have been made for cleanup and maintenance activities with radiation fields measured to 50 rad/hr beta-gamma for brief periods. External exposure continues to accumulate from the decontamination facility [49].

2.3.6 Fluorinel Dissolution Process and Fuel Storage Facility, 1984 to Present

FAST has two parts: a spent fuel storage area and the FDF. The storage area consists of six stainless-steel-lined pools for spent nuclear fuel storage. The FDF includes a hot cell with 6-ft-thick

concrete walls where spent fuel was dissolved in an acid solution. The FDF was used on Zircaloy-clad naval fuels and used three batch-processing dissolver-complexer trains. Soluble neutron poisons and limitations on mass provided criticality control. When reprocessing ended in 1992, uranium and hazardous materials were flushed from the FDF, and this part of the facility was placed in a standby condition. About 1,546 kg of uranium were reprocessed using the Fluorinel dissolution process (Knecht et al. 1997, p. 8).

Internal exposure potential existed from airborne radioactivity that might have occurred from the processes associated with the FAST [50].

External exposure occurred from the unloading and loading of irradiated fuel, from underwater fuel element examination, and from work in the FDF hot cell environment, which created radiation exposure as high as 50 rad/hr for brief periods [51].

2.3.7 INL Comprehensive Environmental Response Compensation and Liability Act Disposal Facility Complex, July 2003 to Present

The INL Comprehensive Environmental Response, Compensation, and Liability Act Disposal Facility (ICDF) Complex is a new facility south of INTEC and adjacent to the existing percolation ponds. It is designed and authorized to accept wastes from INL actions under the act. The Complex includes the necessary subsystems and support facilities to provide a complete waste management system. The major components include disposal cells (landfill), an evaporation pond (consisting of two cells), and the Staging, Storage, Sizing, and Treatment Facility. The Complex covers approximately 40 acres, with a landfill disposal capacity of approximately 510,000 yd^3. The evaporation pond is designated as a RCRA Corrective Action Management Unit and is the disposal site for leachate and other aqueous wastes from operating the Complex. In addition, other aqueous wastes such as existing purge water can be disposed of in the evaporation pond in accordance with the ICDF evaporation pond waste acceptance criteria.

Internal exposure from airborne radioactivity would exist if the integrity of the packaged material was compromised during the handing and storage process [52].

External exposure occurs when shipments of MFPs and MAPs are placed in storage at the facility. Normal radiation levels are not permitted to be greater than 200 mrem/hr at any edge of the transporting vehicle. Higher radiation fields might be permitted, under special conditions, by proper management authority [53].

2.3.8 TMI-2 Independent Spent Fuel Storage Installation, CPP 1774, 1998 to Present

The Independent Spent Fuel Storage Installation (ISFSI) is a new U.S. Nuclear Regulatory Commission (NRC)-licensed dry storage area for spent fuel and debris from the TMI accident. Fuel and debris were transferred to the INL TAN for examination, study, and storage after the accident. The fuel and debris were transferred to the ISFSI, which provides safe, environmentally secure, aboveground storage in metal casks inside concrete vaults. The transfer was completed in mid-2002.

Internal exposure potential could exist from leaks or off-gassing from the storage containers [54].

External exposure occurred during the unloading and placement of fuel in the storage vaults [55].

2.4 ARGONNE NATIONAL LABORATORIES–WEST, FEBRUARY 1951 TO PRESENT

ANL-W was originally known as "the Idaho Division" of ANL. ANL-W is an extension of ANL-E near Chicago. ANL is a DOE research laboratory operated by the University of Chicago. The original INL site is in the southwest portion, approximately 18 mi via Highway 20/26 east of Arco, 40 mi via Highway 26 northwest of Blackfoot, or 50 mi west via Highway 20 from Idaho Falls. The original ANL-W location is now a National Historic Landmark and it is the site of the now-decommissioned EBR-I, Boiling Water Reactor (BWR) Experiment, the Argonne Fast Source Reactor (AFSR), and the Zero Power Reactor 3 (ZPR-III) facilities.

The present ANL-W site is in the southeast portion of INL, about 35 mi west of Idaho Falls. There are 52 major buildings at this site, including reactor buildings, laboratories, warehouses, technical and administrative support buildings, and craft shops. In February 2005, the ANL-W was merged into the INL, and was subsequently renamed the Materials and Fuels Complex (MFC). However, for convenience, this TBD uses ANL-W because it has been this area's name for most of its history. It is also the name that it is known as by most of the EEOICPA claimants.

2.4.1 Experimental Breeder Reactor No. I, August 24, 1951 to December 30, 1963

EBR-I, the first reactor built at INL, was a NaK-cooled, solid-fuel (enriched uranium), unmoderated, heterogeneous fast reactor designed for full-power operation at a level of 1 MW. It was built to explore the possibilities of breeding nuclear fuel and for the use of liquid metal cooling. A blanket of ^{238}U around the core provided the fertile material in which nuclear material breeding took place. Because the primary coolant was intensely radioactive during and shortly after operation, all primary components were enclosed in concrete-shielded cells. The secondary coolant, which was nonradioactive, required no shielding (Kittel, Novick, and Buchanan 1957). The facility was entirely inside a single building of brick, concrete, and steel. Construction on EBR-I began in May 1949 and was complete in April 1951. Reactor startup occurred on August 24, 1951. (Stacy 2000, p. 276)

On November 29, 1955, the reactor suffered a 40% to 50% core meltdown. Radiation detection instrumentation measured radioactivity in the building above normal background levels, and all personnel were evacuated. After the partial meltdown, the core assembly was removed from the reactor using a temporary cave on the reactor top and shipped to ANL- E (Kittel, Novick, and Buchanan 1957). The core was replaced and the reactor remained operational until December 30, 1963. On August 26, 1966, EBR-I was dedicated as a National Historic Landmark. The principal radiological activity associated with the coolant during operation and shortly after shutdown was ^{24}Na with a half-life of 15 hours. The saturation level at full-power operation was approximately 24 µCi/g. The second most significant activity was about 2 µCi/g of ^{137}Cs, which apparently entered the system during and after the meltdown incident. No other long-lived activity was identified in the primary coolant. Short-lived activity in the form of ^{133}Xe and ^{135}Xe was observed in the cover gas (Haroldsen et al. 1963).

Internal exposure potential existed from airborne radioactivity from mixed fission products and activation products [56].

External exposure was received by personnel from MFPs and activation products during activities associated with reactor operation and maintenance [57].

2.4.2 Boiling Water Reactor Experiment No. 1, May 1953 to July 22, 1954

The date information in this discussion is from Stacy (2000, pp. 144, 274). BORAX-I was an open-top, water-cooled, water-moderated, BWR used to conduct a series of nondestructive experiments in the latter part of 1953 and early summer of 1954 (Dietrich 1956). The reactor was built in an excavated area of earth approximately one-half mile northwest of EBR-I and was housed in a 10-ft-diameter tank open to the atmosphere. The control room was approximately one-half mile away near the EBR-I reactor.

BORAX-I was intentionally destroyed in its final experiment on July 22, 1954. The explosion scattered fuel plate fragments and other debris over an area of approximately 200 ft by 350 ft south of the reactor area fence. Instrumentation at the control center showed an instantaneous radiation level higher than 500 mrem/hr, which decreased in about 30 s to 25 mrem/hr and within 5 min to less than 1 mrem/hr. A detailed discussion of this incident is available (Griffiths, Sill, and Wilhelmsen 1956; Brodsky and Beard 1960).

Internal exposure might have occurred from airborne radioactivity during operation and other activities associated with a BWR, the core destruction, coolant, and cleanup activities [58].

External exposure occurred from direct radiation associated with the reactor operation maintenance activities from MFPs and activation products [59].

2.4.3 Boiling Water Reactor Experiment No. 2, October 19, 1954, to March 1955

The date information in this discussion is from Stacy (2000, p. 275). BORAX-II was designed and built to replace BORAX-I to investigate a new reactor that would more closely approximate the characteristics of a practical power reactor operating on the boiling-water principle (Dietrich 1956). BORAX-II, a short distance northeast of the BORAX-I site, was built in 1954 and became operational on October 19, 1954. BORAX-II was significantly larger than BORAX-I. The vessel was shielded by concrete and housed in a sheet-metal building. Tests of new core combinations were tried using varying enrichments of ^{235}U in metal fuel plates (AEC 1962). It was a boiling-water system operating at 300 psi, making it essentially a power experiment. The power level was about 6.4 MW (t) but, because it had no turbine generator, it produced no electricity. The energy produced was released in the form of steam (Dietrich 1956). In 1955, a turbine generator was added to BORAX-II and testing was done to demonstrate that turbine contamination would not be a significant problem in BWRs.

Internal exposure was possible from MAPs associated with work activities from the reactor coolant [60].

External exposure occurred during routine operations and with loading and unloading of the reactor fuel [61].

2.4.4 Boiling Water Reactor Experiment No. 3, June 9, 1955, to 1956

The date information in this discussion is from Stacy (2000, p. 275). As a result of the BORAX-I and -II tests, a program began in March 1955 to modify the BORAX-II reactor to produce electricity. The modified reactor became BORAX- III. The previous BORAX reactors were not designed to produce electricity, so a turbine generator was added to the facility to convert thermal energy to electricity. The modified facility was capable of generating 12 MW of thermal energy and 2,300 kW of electrical energy. For 2 hr on July 17, 1955, BORAX-III generated approximately 2,000 kW of electricity; 500 kW were used to power the BORAX-III facility, 1,000 kW were used to power the CFA, and 500 kW

were used to light the entire town of Arco, Idaho (Stacy 2000, p. 275). BORAX-III became the first nuclear power plant in the world to generate electricity for an entire city.

Internal exposure was possible from MAPs associated with work activities from the reactor coolant [62]. Fuel cladding failure occurred, and the resulting radionuclides were identified in the reactor water (Zinn et al. 1956).

External exposure occurred from routine operation and loading and unloading of the reactor [63]. High radiation levels (up to 430 mR/hr) were reported associated with the steam system. Decay curves were measured for the steam, condensate, and reactor water (about 1×10^4 higher activity) (Zinn et al. 1956). Nitrogen-16 was identified as the principal source in the coolant (Dietrich, Lichtenberger, and Zinn 1956).

2.4.5 Boiling Water Reactor Experiment No. 4, December 3, 1956, to June 1958

The date information in this discussion is from Stacy (2000, p. 275). BORAX-IV, the successor to BORAX-III, began operation in December 1956. This reactor, with a design power of 20 MW (t), was used principally to test high-thermal-capacity fuel elements made from ceramics of uranium and thorium (Handwerk, Hoenig, and Lied 1957). Like the previous BORAX reactors, BORAX-IV was a BWR operating at 300 psig. It was capable of producing 2.5 MW of electricity. It was brought to criticality on December 3, 1956, at atmospheric pressure. It operated with a core of uranium-thorium fuel elements until April 17, 1957. Beginning in May 1957, it was operated with a 59-element core at 300 psig and 216°C and continued intermittent operations until December 5, 1957. After the core was revised to increase the maximum power, the reactor was restarted on February 19, 1958, to evaluate the effect of operating with a fuel element defect and to locate defective elements in the core (Robertson and Hall 1959). BORAX-IV released approximately 4,565 Ci of short-lived radionuclides to the atmosphere in March 1958 (Novick 1958). It operated until June 1958. The following measurements were made during reactor operations (Robertson and Hall 1959):

1. Radiation levels of the steam plant equipment
2. Quantitative determination of fission gases ^{138}Xe and ^{88}Kr, which were released to the atmosphere through the air ejector
3. Analysis of reactor water, condensed steam before the turbine, and condensed steam after the turbine (hot-well condensate) for fission products
4. Area contamination downwind from the reactor

Internal exposure might have occurred during work with the defective fuel elements or during planned releases of short-lived fission activity and from airborne MAPs/MFPs associated with the reactor coolant [64].

External exposure occurred during operation and work with loading and unloading of the reactor [65].

2.4.6 Boiling Water Reactor Experiment No. 5, February 9, 1962, to September 1964

The date information in this discussion is from Stacy (2000, p. 275). BORAX-V was a flexible BWR with the same configuration as BORAX-IV, used primarily for testing nuclear superheating concepts. The facility was operational from February 9, 1962, until September 1964.

Internal exposure might have occurred from coolant and airborne activity during routine BWR operation with fuel elements made of ceramics of uranium and thorium, and associated maintenance work [66].

External exposure occurred from routine activities associated with reactor operation and maintenance [67].

2.4.7 Zero Power Reactor No. 3, October 1955 to November 1970

The date information in this discussion is from Stacy (2000, p. 282). ZPR-III was a low-power, split-table reactor that achieved criticality by bringing two halves of a fuel configuration together. It was used to determine the accuracy of predicted mass geometries and critical measurements for fast reactor core designs.

The building consisted of a reinforced concrete high bay assembly room and a one-story section containing the control room, work room, vault, laboratory rooms, offices, etc. The assembly (reactor) room of reinforced concrete was approximately 45 ft by 42 ft by 29 ft high (Brittan et al. 1961).

The assembly machine was a platform on which two tables or carriages were mounted, one of which was moveable. Half of the reactor was built up on each carriage by inserting drawers containing the reactor material into a matrix structure. Each half of the assembly contained five safety control rods and a 15-Ci polonium-beryllium neutron source. A hinged platform could be swung into place between the halves on which workers could stand while loading or unloading the machine.

The storage vault room was approximately 29 ft long by 26 ft wide with walls and roof of reinforced concrete 9 in. thick. The fuel slugs were stored either on racks or in special “birdcage”-type containers that limited the storage density to 2 kg of ^{235}U per cubic foot (Brittan et al. 1961).

Internal exposure potential existed from possible airborne radioactivity during maintenance operations from MFPs/MAPs [68].

External exposure occurred during maintenance activities and during loading and unloading of the reactor fuel [69].

2.4.8 Argonne Fast Source Reactor, October 29, 1959, to Late 1970s

The date information in this discussion is from Stacy (2000, p. 274). FSR was a small fast reactor facility designed to produce neutrons for the development of special equipment for the fast reactor programs of EBR-I, EBR-II, and ZPR-III. The reactor, with a design power of 1 kW, was in a prefabricated Butler-type building with its own heating and air compressor plant built in 1958 near the EBR-I facility. No water was plumbed into the building. Control and safety mechanisms were in a pit below the reactor. The reactor, designed to supply both fast and thermal neutron fluxes for laboratory experiments, was built around a cylindrical core of solid HEU with a blanket of solid DU (Brunson 1959, pp. 8-9). Reactor startup occurred on October 29, 1959; the reactor was operational until sometime in the late 1970s when it was moved to a new location adjacent to the ZPPR facility at the ANL-W site. The reactor is now shut down and defueled.

Internal exposure might have occurred during routine operations that could create airborne radioactivity [70].

External exposure occurred from maintenance activities and loading and unloading of fuel [71].

2.4.9 Transient Reactor Test Facility, February 23, 1959, to April 1994

The date information in this discussion is from Stacy (2000, p. 282). TREAT was an air-cooled thermal heterogeneous system designed to evaluate reactor fuels and other material under conditions simulating various types of reactor excursions. Construction began in February 1958 and ended in November 1958, and criticality was achieved on February 23, 1959. The TREAT complex consists of a reactor building and a control building approximately 1 mi northwest of the EBR-II containment building (Freund et al. 1960).

The reactor building features a high bay section and an adjacent service wing. The high bay section contains the reactor, fuel storage pit, instrument room, and the basement subreactor and equipment rooms. The control building is a single-story concrete block structure approximately one-half mile from the reactor that contains control panels and necessary instrumentation for remote control of the reactor.

Shielding permitted personnel access around and on top of the reactor during steady-state operations at 100 kW. Access to the subreactor room was controlled during steady-state operation. Before transient operations, the building was evacuated. General neutron and gamma radiation levels 10 ft from the reactor during operations at 100 kW were (Freund et al. 1960):

- Fast neutron Negligible
- Thermal neutrons 50–1,500 n/cm^2/s
- Gamma 5–8 mrem/hr

Internal exposure might have occurred during routine operations that could create airborne radioactivity; however, it was not expected to occur [72].

External exposure occurred from routine operations [73].

2.4.10 Experimental Breeder Reactor No. II, September 30, 1961 to September 30, 1994

The date information in this discussion is from Stacy (2000, p. 276). EBR-II, at the ANL-W site, is a liquid sodium-cooled, unmoderated, heterogeneous fast breeder reactor rated at 62.5 MW (t), with an intermediate closed loop of secondary sodium and a steam plant capable of producing electric power through a conventional turbine generator. A fuel processing facility is attached to the reactor. EBR-II was designed to prove the breeding of fuels, the feasibility of a central power station, and onsite fuel processing. These objectives were met in the late 1960s, and the role of EBR-II changed to test reactor. Construction of EBR-II ended in May 1961, and the reactor reached criticality on September 30, 1961. It operated until September 30, 1994, when it was taken to a subcritical configuration and shut down to start a defueling operation. On January 19, 2001, ANL-W verified that the liquid-metal sodium coolant had been completely drained from the reactor vessel. At present, the reactor is defueled, the sodium systems have been drained, and the power plant is depressurized.

The reactor was submerged in a primary tank containing approximately 90,000 gal of liquid sodium. This tank was suspended in an airtight steel-shell containment building of 1-in.-thick steel plate, which would contain an accidental release of fission products, etc., from the primary system. The structure of the primary system is designed to contain the energy release associated with a reactor incident. The reactor building is designed to confine the effects of a maximum sodium-air interaction caused by a major sodium release. The reactor consists of an enriched core surrounded on all sides by a fertile blanket of depleted uranium (McVean et al. 1962; Koch et al. 1957).

The Sodium Plant contains the pumping, purification, and storage facilities for the secondary sodium system. It also contains a receiving station for the sodium. The building was not normally occupied. The primary and secondary coolant from EBR-II was converted in the Sodium Processing Facility from its elemental, chemically unstable form, to a chemically stable composition suitable for landfill disposal.

The Fuel Manufacturing Facility (FMF) is a secure facility designed for the fabrication of EBR-II fuel. The FMF vault stores special nuclear material in support of the EBR-II shutdown.

An additional building, the Laboratory and Office Building near the EBR-II plant, provided supporting analytical and personnel facilities.

2.4.11 Hot Fuel Examination Facility, 1964 to Present

The Hot Fuel Examination Facility (HFEF) complex comprises two hot cell facilities, HFEF/South and HFEF/North. HFEF/South, originally known as the Fuel Cycle Facility and/or the Fuel Conditioning Facility, was used to demonstrate remote processing and refabrication of uranium-fission, metal-alloy driver fuel elements in a closed cycle with EBR-II.

Some 35,000 fuel elements were remotely reprocessed and refabricated into EBR-II subassemblies between 1964 and 1968. HFEF/South contains two large heavily shielded hot cells, one with an inert gas (argon) atmosphere, and the other with an air atmosphere. The shielding walls of both cells are of high-density concrete. The HFEF/South air cell was decontaminated and refurbished in 1969 and again in 1976 (Baca 1979).

HFEF/North is a large alpha-gamma hot cell facility that was activated in March 1975. This facility provided the capability for postirradiation and nondestructive or destructive examination of fuel and material experiments that were irradiated in EBR-II (Baca 1979). HFEF/North contains two hot cells, one with an argon gas atmosphere and the other with an air atmosphere. The air atmosphere cell was known as the decontamination cell. The shielding walls of both cells are of high-density concrete (Baca 1979). HFEF began operation as a fully automated facility for examining highly radioactive experimental reactor fuel elements and other components in 1975. The examinations in HFEF provide data that are essential for determining the performance and condition of fuels and materials that are irradiated in DOE reactor facilities. HFEF continues in operation as a vital component of the DOE energy research program.

Remote characterization of material to be shipped to the Waste Isolation Pilot Plant (WIPP) in New Mexico for disposal takes place in the Waste Characterization Area of the HFEF high bay.

Internal exposures might have occurred during cell entries, when suspended radioactive contamination materials could cause airborne radioactivity from MFPs and MAPs [74].

External exposure occurred when entries to the hot cell were made after experiment processes or during equipment maintenance and refurbishment [75].

2.4.12 Zero Power Physics (Plutonium) Reactor, April 18, 1969 to April 1992 (Standby)

The date information in this discussion is from Stacy (2000, p. 282). ZPPR is a split-table critical facility approximately 300 m from EBR-II in the ANL-W area and about 3 mi north of U.S. Highway 20. The facility is divided into two areas, the mound area and the support wing. The mound area consists of the reactor cell, fuel storage vault, workroom, and equipment rooms as well as access and escape

tunnels. The reactor cell is a 50-ft-diameter circular room with floor and walls of reinforced concrete. The roof consists of layers of washed and dried sand and gravel that is supported by a catenary cable network.

The basic element of the ZPPR is a bed-and-table system, which holds the matrix assembly. The two tables, one moveable and one stationary, are supported on a cast-steel bed. Neutron fields from the plutonium fuel were present between the two halves when the reactor was off and open (Simons, Young, and Thalgott 1972). The control and safety rod drives were mounted near the rear of each table. The main floor consists of the reactor control room, offices, an electronics shop, and a core coating room. The core coating room, adjacent to the control room and the entrance to the mound area, was used to clean core stimulants such as ^{235}U and stainless steel. The room contains two hoods for handling suspect materials and a core coating machine that was used primarily to dry and coat DU with a protective film.

Internal exposure potential was minimal due to the use of hoods and other protective equipment [76].

External exposure occurred from working with reactor processes, loading and unloading fuel, etc. [77].

2.4.13 Neutron Radiography Facility, October 1, 1977, to Present

The Neutron Radiography (NRAD) Facility is a 250-kW, steady-state Training, Research, Isotopes, General Atomics reactor in the basement beneath the HFEF/North main cell. The reactor core consists of fuel elements that are surrounded by graphite assemblies. The core is submerged in a water-filled tank. NRAD began operation in March 1978 with two radiography stations. The East station services the hot cell complex where specimens can be radiographed without removing them from the hot cell environment. The North station is outside the cell in a separate, clean, shielded location for the radiography of irradiated or unirradiated items without introducing them into the contaminated cell. Cask handling and specimen shielding allow for full-size reactor assemblies. The radiography room is easily accessible for development work (Richards and McClellan 1979).

NRAD has limited irradiation capabilities in the core. It has a water-filled port at the center of the core and a dry port at the edge of the core. NRAD operates an MF Physics linear particle accelerator that is used for nondestructive assays of waste and expended nuclear fuel.

Internal exposure potential exists from possible airborne radioactivity, primarily from the hot cell environment from MFPs and MAPs [78].

External exposure occurred during sample handling and maintenance that is associated with radioactive samples. Remote handling techniques are used to minimize dose [79].

2.4.14 Fuel Assembly and Storage Building, 1970 to Present

The Fuel Assembly and Storage Building is a multipurpose facility that supports development of low-enrichment uranium fuel for research reactors, storage of spent fuel, and examination of the condition of other experimental projects. The East (clean) room houses offices, restrooms, and other support facilities. The West room contains a vault for the storage of nuclear material. It also contains equipment for performing materials testing and preparing metallurgical samples and inert atmosphere gloveboxes and hoods. The facility ceased fuel assembly in about 1990, but other radiological work is ongoing.

Internal exposure might occur from airborne radioactivity from the described processes from the uranium fuel and spent fuel examinations [80].

External exposure occurred from the movement of radiological samples and reactor fuel [81].

2.4.15 Other Argonne National Laboratory-West Support Facilities

At ANL-W, an Analytical Laboratory provides the capability for performing chemical and physical measurements of radioactive and nonradioactive samples. This facility includes six analytical hot cells (the Junior Cave area) and general and specialized chemistry laboratories. Personnel were subjected to radiation levels in the multiroentgen-per-hour range on occasion in the Junior Cave area.

The Radioactive Scrap and Waste Facility provides in-ground retrievable dry storage for nuclear fuels and other highly radioactive scrap and waste and interim storage for EBR-II spent fuel.

The Radioactive Liquid Waste Treatment Facility evaporates low-level radioactive liquid waste from ANL-W facilities into solidified residue that is packaged in shielded containers.

The Industrial Waste Pond is an unlined evaporative seepage pond that is fed by a system of drainage ditches. It has been used since 1964 to receive wastewater from a number of sources. The largest sources of liquid industrial waste that go to the Industrial Waste Pond are blowdown effluents from the main and auxiliary cooling towers, auxiliary boilers blowdown, water from once-through air conditioning, and cooling water from other sources. There might be inadvertent low-level radioactive contamination in this pond.

The three sanitary Sewage Treatment Ponds north of EBR-II cover an area of about 2 acres. These ponds are not suspected of containing radiological contamination.

Internal exposure is possible from airborne radioactivity from the types of samples worked in these facilities [82].

External exposure occurred from working with the variety of radioactive materials in these facilities (MFPs, MAPs, and TRU materials) [83].

2.5 RADIOACTIVE WASTE MANAGEMENT COMPLEX, MAY 1952 TO PRESENT

The RWMC is 51 mi west of Idaho Falls. The first 13 acres were fenced in May 1952 for shallow-land disposal of solid low-level radioactive waste and burial of TRU waste and hazardous substances such as organic and inorganic chemicals. The current RWMC mission includes interim storage of TRU waste and shipment of stored TRU waste to the WIPP near Carlsbad, New Mexico, for permanent disposal.

For the first 2 years, only low-level radioactive waste was buried at the RWMC. In 1954, the Rocky Flats Plant (RFP) in Colorado began shipping defense waste with TRU elements. By 1957, the original 13 acres were nearly filled, and the RWMC was expanded to 97 acres. In 1970, it was expanded again to 168 acres and is currently 177 acres. After 1970, TRU waste was placed in retrievable storage on asphalt pads and covered with an earthen berm. This waste is stored in drums and boxes in engineered modules. From 1970 to the present, low-level waste has been disposed of in 20 pits, 58 trenches, and 21 soil vault rows. INL has been repackaging (as needed) and shipping TRU waste to the WIPP (Stacy 2000, pp. 88-99; Lenhard et al. 2004, pp.3-4; Till et al. 2002, pp. 26-29).

Early packaging configurations were thought to be unsuitable for extended storage and could present future hazards to the workers, the public, or the environment. Early wastes were pushed out of trucks into open pits or trenches and covered with soil by heavy equipment, which might have occasionally damaged containers in the covering process. Long-tongued dump trailers were used to minimize exposure to personnel from the wastes. Other types of heavy equipment such as cranes and cherry pickers were used to pick shielded containers from trucks for unloading the contents into soil vaults, etc. (Stacy 2000, pp. 88-99; DOE 1995, pp. 26-29; Till et al. 2002, pp. 26-29) There were occasional brief exposures greater than 100 mrem/hr (photon energy ≥250 keV).

Radionuclides at the RWMC cross the spectrum based on the character of operations at this facility. Those listed in Table 2-3 would represent many of the long-lived fission products. MAPs from reactor facilities were also a concern for external exposure.

In addition to administrative buildings, the RWMC has the facilities and processes that are described in the following sections.

The following exposure potentials are for all subsections of Section 2.5.

Internal exposure might have resulted when workers were close to breached and or leaking waste containers that contained MFPs, MAPs, TRU materials, etc., from the materials in storage at the RWMC [84].

External exposure might have resulted from working adjacent to the waste containers in storage or disposal in the form of intermediate-level, low-level, TRU, and mixed waste and from Stored Waste Examination Pilot Plant X-ray (30 to 450 keV) and neutron (2 to 20 MeV) waste package examination and certification processes. Approximately 150 Department of Transportation (DOT) 6M drums that contain as much as 500 g of $^{232/233}$U per drum have been stored under earthen covers on pads in the Transuranic Storage Area (TSA) [^{232}U is always present as a contaminant with ^{233}U and accounts for a significant in-growth of high-energy gamma emitters]. Depending on the waste type, radiation exposure levels might have been near background levels or could have exceeded a few hundred millirem per hour with photon energies greater than 250 keV for short periods [85].

2.5.1 Subsurface Disposal Area

The 97-acre Subsurface Disposal Area is in the western section of the RWMC. It contains an active shallow-land burial area for the permanent disposal of solid low-level waste. It also contains pits and trenches where mixed TRU and low-level waste was buried between 1954 and 1970. Solid waste from RFP comprised a large fraction of the waste that has been received at the RWMC. For example, in 1969 approximately 250,000 ft^3 of waste from RFP with a reported activity of more than 35,000 Ci was buried at the RWMC. RFP waste was usually contaminated with plutonium isotopes and ^{241}Am (Till et al. 2002, pp. 26–27). The total activity of INTEC waste buried at this area is 2.8E+05 Ci. Of this, 98% is represented by three radionuclides: Co-60 (57%), Sr-90 (20%), and Cs-137 (21%). Transuranic radioactive waste amounts to only about 100 Ci (Vail, Carboneau, and Longhurst 2005, p. 44).

2.5.2 Intermediate Level Transuranic Storage Facility

The Intermediate Level Transuranic Storage Facility has had 53 drums of ^{233}U stored in metal cargo containers in an open yard that is surrounded by concrete block shielding. The facility also has several in-ground shielded storage wells that are used to store highly radioactive materials.

2.5.3 Transuranic Storage Area

The 56-acre TSA is in the southern section of the RWMC and is dedicated to storage of contact-and remote-handled packages of solid TRU waste. This waste was received at INL after 1970 and was stored aboveground (DOE 1995, p.27).

2.5.4 Stored Waste Examination Pilot Plant

The Stored Waste Examination Pilot Plant certifies waste to ensure that it meets repository acceptance criteria. Examinations use a shielded 450-kVp X-ray facility and a neutron assay system that develop 14-MeV neutrons. Certified waste has been stored temporarily in permitted storage areas until it is packaged and sent to the WIPP. Uncertified waste is segregated until processing is available that will enable it to meet acceptance criteria (DOE 1995, p.27).

2.5.5 Transuranic Package Transporter Loading Station

This loading station is used to load TRU waste into Transuranic Package Transporter-II shipping containers for shipment to the WIPP. Responsibility for the station has been turned over to British Nuclear Fuels Limited.

2.5.6 Advanced Mixed Waste Treatment Project

British Nuclear Fuels Limited operates the Advanced Mixed Waste Treatment Project under contract with DOE. The facility will retrieve and process approximately 65,000 m^3 of mixed TRU waste in temporary storage at the TSA, treat the waste to meet environmental laws and disposal criteria and package it for shipment to the WIPP (NRC 2009, p. 206).

2.6 CENTRAL FACILITIES AREA, AUGUST 2, 1943, TO PRESENT

The CFA is the main service and support center for INL programs and the other technical areas on the 890-mi^2 site. It is about 50 mi west of Idaho Falls, Idaho, just north of Highway 20 in the south-central portion of the site. CFA buildings and activities support transportation, maintenance, capital construction, environmental and radiological monitoring, security, fire protection, warehouses, calibration laboratories, and a cafeteria. There is a small amount of research and development work. What is now the INL started as an offshoot of the Naval Proving Ground command area (dedicated August 2, 1943), where the Navy tested ordnance for fighting ships. Buildings the Navy constructed became the staging area for INL development that began in earnest in 1950. The area continued to expand as a central service area for INL. Eventually it acquired the name Central, or officially CFA (Stacy 2000, pp. 55-57).

In the late 1950s and into the 1960s, small amounts of radioactivity were processed through a Sewage Treatment Plant, CF-674, to a drying pond. Most of the radioactivity was from the hot laundry, although small amounts could enter from CF-656 and CF-690. CF-656 was a Reactor Engineering Laboratory where tracer-level radionuclide and chemistry work occurred. CF-690 included the laboratory where analytical chemistry was done on bioassay samples, naturally occurring radionuclides, and other special projects. The dispensary included an X-ray unit for medical use (30 to 250 keV) [86].

The following exposure potentials are for all subsections of Section 2.6.

Internal exposure. CFA internal exposure potential was primarily from MFPs, MAPs, and TRU materials from articles that were cleaned at the laundry facilities. Laundry facility workers were included in bioassay programs [87].

External exposure. CFA external exposure potential is greatest from calibration sources and X-ray equipment at the Health Physics Instrument Laboratory (HPIL) and the DOELAP Irradiation Facility with photon energies greater than 250 keV from calibration sources and 30 to 250 keV for X-ray photons. Neutron energies range from 2 to 20 MeV from the ^{252}Cf source and the AmBe source. Personnel who work in radiological areas must wear dosimetry devices [88].

2.6.1 Hot Laundry, 1950 to 2002

The laundry, in the east portion of the CFA, washed coveralls and other protective clothing items that were used in radiological work. The laundry drain went to a septic tank and drain field with other sanitary waste. The laundry facility and drain field(s) are sources of low-level radioactive contamination, which covers the spectrum inherent to work in radiological contamination areas. The old laundry facility (CF-699) was used from 1950 and was demolished in 1994. The new laundry facility (CF-617) was used from 1981 to 2001 and demolished in 2002 (Rockhold 2007).

2.6.2 Health Physics Instrument Laboratory

The HPIL, CFA-633, was a calibration facility for radiological instrumentation standardization. The HPIL used ^{252}Cf neutron, alpha, beta, and gamma sources for health physics instrument calibrations. All the sources were sealed [89]. A new facility, CF-1618, was completed in late 2002 and includes six automated irradiator systems and provides expanded neutron, gamma, and X-ray irradiation capabilities. The higher level sources require external exposure control, and personnel in the radiological work area must wear applicable dosimetry devices [90].

2.6.3 DOE Laboratory Accreditation Procedure Irradiation Facility

The DOELAP Irradiation Facility, CF-636, is an aboveground shielded bunker that houses radioactive sources the dosimetry branch uses for radiation measurement equipment calibrations. The bunker is on the access road east of the main road into the CFA. It houses an X-ray facility, seven 1-Ci ^{241}Am sources, beta sources, and two ^{137}Cs sources (20 and 1,000 Ci). At one time, there was an AmBe neutron source at this facility [91].

2.6.4 Radiological and Environmental Sciences Laboratory

The Radiological and Environmental Sciences Laboratory, which DOE operates in CF-690, evaluates low-level environmental and other laboratory samples that pose minimal radiological risk from internal or external pathways [92]. The dosimetry facility, also in CF-690, has been used to process external dosimetry devices such as film, thermoluminescent dosimeters, etc.

2.6.5 CF-674 Building, 1953 to 1969

CF-674 was used from 1953 to 1969 as a Chemical Engineering Laboratory to conduct calcine experiments on simulated radioactive waste. The experiments created liquid waste streams that were discharged to the CFA-04 pond. This waste stream was contaminated with calcine that contained low-level radioactive waste (DOE 2003).

2.7 TEST REACTOR AREA, MARCH 31, 1952, TO PRESENT

The TRA is approximately 5 mi north of the CFA. Eight reactors have been built and operated in the TRA. Three of the reactors – MTR, Engineering Test Reactor (ETR), and Advanced Test Reactor (ATR) – were high-flux reactors for materials testing. The remaining five – Reactivity Measurement Facility (RMF), Advanced Reactivity Measurement Facility No. 1 (ARMF-1), ARMF-2, Engineering Test Reactor Critical (ETRC), and Advanced Test Reactor Critical (ATRC) – were low-power reactors for reactivity measurements. At present, only the ATR and ATRC are operational (Stacy 2000, p. 274).

Other TRA facilities of radiological concern are the TRA Hot Cells, Gamma Facility, Radiation Measurements Laboratory, Radiological Chemistry Laboratory, Liquid Waste Disposal Ponds, and High-Level Liquid Waste Disposal Tanks and Transfer Facility. When DOE merged INL and ANL-W in 2005, TRA became the Reactor Technologies Complex.

All personnel who enter the TRA must wear a dosimetry badge, and those who work in or near radiological control areas must wear pocket ionization chambers. Personnel who work in radiological control areas are on a routine bioassay program and receive routine whole-body counts.

2.7.1 Materials Test Reactor, March 31, 1952, to April 23, 1970

The date information in this discussion is from Stacy (2000, p. 278). The MTR (TRA 603) was the original reactor at the TRA and the second reactor to be operated at INL. This water-cooled and water-moderated reactor used enriched uranium fuel and was a key part of the AEC postwar reactor development program. It operated at a power level of 30 MWt until September 1955, when output was increased to 40 MWt. It supplied a high neutron flux in support of a reactor development program that irradiated potential reactor fuels and structural materials (Stacy 2000, pp. 278-279).

The MTR gave researchers several options to achieve sample irradiation. Lead experiments entered from the top of the reactor with positions around the core. Pneumatic ports on the reactor top enabled the insertion of capsules for irradiation in the graphite region around the core. A hydraulic rabbit system underneath the reactor enabled the insertion of specimens and their discharges to the canal during reactor operation. In addition, horizontal and angular beam holes made it possible to perform cross-section measurements and other physics research including several neutron experiments. The high-flux radiation fields available in this reactor made it possible to accelerate the screening of test materials. In its early years, the MTR contributed to the design of pressurized-water, organic-moderated, liquid-metal-cooled, and other reactors. Its successful operation resulted in a family of plate-type reactor fuels.

The MTR logged more than 125,000 operating hours and more than 19,000 neutron irradiations. During August 1958, it became the first reactor to operate using ^{239}Pu as fuel at power levels as high as 30 MW. In early 1970, the MTR was again fueled with ^{239}Pu. The last core was named Phoenix after the legendary bird that lived 500 years, burned itself to ashes, then rose to live again. The plutonium cores demonstrated that a plutonium-fueled, water-moderated reactor could be controlled satisfactorily. In August 1970, the MTR was again brought to power for a 24-hour run to irradiate 1,000 biological samples for iodine analysis (Stacy 2000, pp. 209-210, 278-279) [93].

Internal exposure was most probable during the first few hours of shutdown. When the reactor top was removed, airborne fission products would be released. During shutdown, airborne radioactivity of MFPs and MAPs from maintenance activities resulted in the potential for internal exposure. Some experiments in loops resulted in releases, particularly of MAPs [94].

External exposure occurred during the sampling of a test reactor and the associated maintenance activities. The major contributors to external exposure were MFPs and MAPs that emitted beta and gamma radiation with energies typically above 250 keV [95].

2.7.2 Engineering Test Reactor, September 19, 1957, to December 1981

The date information in this discussion is from Stacy (2000, p. 276). When the 175-MW ETR started in 1957, it was the largest and most advanced nuclear materials test reactor in the world. It provided larger test spaces than the MTR and a more intense neutron flux. ETR fuel, coolant, and moderator materials were evaluated under environments similar to those of power reactors. Several experimental loop facilities were designed to test the fuels for the ANP Program and the Navy fuel development program.

In 1972, a Sodium Loop Safety Facility was added to the ETR reactor core. With this, the reactor played a new role supporting the DOE breeder reactor safety program. ETR test programs were related to the core design and operation of breeder reactors. As testing progressed, the reactor was modified with a new top closure to accommodate the irradiation loop. Other additions included a helium coolant system and sodium-handling system. The ETR was the first complete reactor facility to be deactivated and the D&D to be documented immediately after shutdown (Stacy 2000, p. 276).

Internal exposure from airborne radionuclides was minimal during normal reactor operation. Exposures might have occurred during shutdown because airborne fission products were often released when the reactor top was removed and access was made to the reactor subpile room and experiment cubicles for maintenance activities. Some releases of activation products from experiments, particularly in loops, occurred [96].

External exposure was received by workers in the reactor area during shutdown and changes of loop and lead experiment samples as required. There were cases of significant gamma fields that exceeded 50 mrem/hr from fission and activation products in the reactor subpile room, loop cubicles, and nozzle trench [97].

2.7.3 Advanced Test Reactor, July 2, 1967, to Present

The date information in this discussion is from Stacy (2000, p. 274). The ATR is the latest materials testing reactor to be built in the TRA. It simulates the environment in a power reactor to study the effect of radiation on steel, zirconium, and other materials. The ATR produces an extremely high neutron flux, which makes it ideal for materials testing. Target materials are exposed to the neutron flux to test their durability in an environment of high temperature, high pressure, and high gamma fields. Data that normally would require years to gather from ordinary reactors can be obtained in weeks or months from the ATR.

The ATR can operate up to a power level of 250 MW. Its unique four-lobed design can deliver a wide range of power levels to nine main test spaces or loops. Each loop has its own distinct environment apart from that of the main reactor core. Smaller test spaces around the loops enable additional tests. In addition, the ATR produces radioisotopes for use in medicine, industry, and other research (Stacy 2000, p. 274).

Internal exposure might have occurred during reactor shutdown from MFPs and MAPs that were released to the air in the occupied environment of the building. The most probable locations would be from the reactor top, experiment cubicles, primary coolant rooms, and subpile room. During reactor operation, areas with airborne radiation are exclusion areas [98].

External exposure is received primarily during reactor shutdown from fission and activation products in the fuel, experiments, and associated hardware. Work on the primary system, in the reactor tank, or in loop cubicles would have associated external exposure. Handling of isotope production samples would produce some exposure dependent on the nuclide and quantity. Typical nuclides include ^{192}Ir, ^{99}Tc, ^{60}Co, and ^{131}I [99].

2.7.4 Reactivity Measurement Facility, February 11, 1954, to April 10, 1962

The date information in this discussion is from Stacy (2000, p. 280). The RMF was a very-low-power reactor in the east end of the MTR canal that operated at a power level of 100 or 200 W. Water was its moderator, reflector, and shield. It was designed to measure reactivity changes in materials irradiated in the MTR or ETR. The RMF was used to assay new and spent fuel elements and to assist in experiment scheduling by evaluating reactivity losses and flux depression caused by in-pile apparatus (Stacy 2000, p. 280).

Internal exposure. There was a potential for internal exposure from MFPs or MAPs from airborne radioactivity during maintenance activities [100].

External exposure was minimal due to the low-power operating level and the depth of the pool. Any external exposures would have come from fission and activation products, primarily during fuel handling [101].

2.7.5 Advanced Reactivity Measurement Facility No. 1, October 10, 1960, to 1974

The ARMF-I reactor was in a small pool in the TRA-660 building east of the MTR building. It was used to determine nuclear characteristics of reactor fuels and other materials for testing in the MTR (Stacy 2000, p. 274).

2.7.6 Advanced Reactivity Measurement Facility No. 2, December 14, 1962, to 1968

The ARMF-II reactor was in the opposite end of the tank occupied by ARMF-I. It had a “readout” system that automatically recorded measurements on data cards. This refinement over the ARMF-I reactor meant operators could process data quickly in computers (Stacy 2000, pp. 274).

Internal exposure from ARMF-I and ARMF-II airborne activity of fission products could exist if a fuel element or sample was damaged during handling [102].

External exposure from ARMF-I and ARMF-II was minimal due to their low operating power levels and the depth of the canal in which they are located. Exposures would be from fission products and/or activation products during fuel or experiment handling [103].

2.7.7 Coupled Fast Reactivity Measurement Facility, 1968 to 1991

When the ARMF-II reactor was modified in 1968, it received a new name, the Coupled Fast Reactivity Measurement Facility. The core was modified to produce a region of high-energy neutron flux to provide physics information about the behavior of fast (unmoderated) neutrons. Physicists studied differential cross-sections and tested calculation methods. The facility contributed to the development of fast neutron reactors (Stacy 2000, p. 275).

Internal exposure potential existed from airborne fission products that might have been released from the fuel [104].

External exposure potential existed at minimal levels from working on the reactor top to move fuel or experimental components. MFPs or MAPs would be the major contributors [105].

2.7.8 Engineering Test Reactor Critical Facility, May 20, 1957 to 1982

ETRC was a full-scale, low-power nuclear facsimile of the ETR in TRA-635, similar in function to the ARMF and ATRC. It was used to determine the nuclear characteristics of fuel and experiments planned for irradiation in ETR and/or the power distribution effects for a given ETR fuel and experiment loading. ETRC enabled operators to predict the nuclear environment when completed experiments were removed or new experiments were added to calculate the irradiation and determine core life, control rod withdrawal sequences, reactivity worth, and core safety requirements.

Mockups of fuel and experiment loadings in ETRC were manipulated until a desired power distribution throughout the core was attained, satisfying pertinent safety requirements. ETRC low-power tests enabled the ETR to operate without interruption, which saved time and money (Stacy 2000, p. 276).

Internal exposure from airborne radioactivity of MFPs from damaged fuel or leaky experimental samples that contained activation products is possible [106].

External exposure potential existed from transfer of irradiated fuel or samples into or out of the ETRC under water to minimize exposure. The operating console was not on the reactor top, which minimized external exposure. External exposure would have been from MFPs and MAPs during loading or unloading fuel or test samples [107].

2.7.9 Advanced Test Reactor Critical Facility, May 19, 1964, to Present

The ATRC performs functions for the ATR similar to those of the ARMF reactors for the MTR. It verified for reactor designers the effectiveness of control mechanisms and for physicists' predictions of power distribution in the large core of the ATR. Low-power testing in the ATRC conserved time so the large ATR could irradiate experiments at high power levels and verified the safety of a proposed experiment before it was placed in the ATR (Stacy 2000, p. 274).

Internal exposure was possible from fission products that were released from damaged fuel or experiment samples that contained activation products. The damage could occur during transfer into or out of the ATRC. Some fuel or samples had been preirradiated in the ATR or other facilities [108].

External exposure occurred during canal work. The operators leaned over the canal parapet to work with irradiated fuel or experiment samples being transferred into or out of the core under water. During reactor operation, access to the canal parapet is restricted. The ATRC canal is 21 ft deep (DOE 1996, p. 9).

2.7.10 TRA Hot Cell Facility, 1954 to Present

The TRA Hot Cells are southwest of the ETR reactor building. They consist of three separate cells with a common operating corridor. The operators are protected by thick concrete walls and special viewing windows, behind which they can handle, photograph, mill, measure, and weigh radioactive samples (Stacy 2000, p. 125). The work in the cells has involved all types of samples including irradiated fuel, TRU materials, and isotope production material. Irradiated samples, including failed reactor fuel, can be put in Cell 1 or 3 from a shielded cask outside the building. Each cell has to be entered periodically to repair equipment or set up for a new job. Entry is through a shielded door in the back of the cell.

Internal exposure might occur during cell entries from the samples and the work with irradiated samples, dust, or particles that could become airborne [109].

External exposure occurs from samples that go in the hot cells. Exposure to beta, gamma, and neutrons has occurred. Fuel samples of different types and different ages result in MFPs, MAPs and TRU materials, including ^{252}Cf. Many samples have been prepared as isotopic sources for industrial or medical applications. Cell entry and sample handling result in most of the external exposure, and some has occurred from sample ports and manipulator removal operations [110].

2.7.11 TRA Gamma Facility, 1955 to Unknown

The TRA Gamma Facility was south of the original TRA main security gatehouse. The facility consisted of a 16-ft-deep canal with cadmium buckets designed to hold spent MTR fuel elements. Experimental samples were inserted in sample tubes and lowered into extremely high gamma fields. Sponsors provided a large variety of materials and samples for gamma irradiation, including food products and some natural substances such as gold, diamonds, and oil. Irradiated samples, which were not radioactive, were surveyed thoroughly for external contamination on removal (Stacy 2000, p. 126).

Internal exposure was possible from a spent fuel element that was used for irradiations that was unlikely to be damaged [111].

External exposure was minimal due to handling procedures and shielding from the canal and transport devices. Fuel was transported in large casks into and out of the canal [112].

2.7.12 Radiation Measurements Laboratory, 1952 to Present

The Radiation Measurements Laboratory (RML) in the MTR west wing was previously called the MTR Counting Room. The RML specializes in measuring quantity and quality of alpha, beta, gamma, and neutron radiation samples. A variety of counting equipment and spectrometers are available in the RML. Over the years, an endless variety of samples has been brought in for counting. Some of the detector shielding was made of pre-World War II battleship steel. The original equipment has been replaced with more modern equipment.

Internal exposure is possible from airborne radioactivity from mishandled samples. The encountered isotopes would include fission products, activation products, TRU materials, and more [113].

External exposure is very low due to the small samples that are required for the counting equipment. Counting room personnel can encounter neutrons and beta and gamma emitters [114].

2.7.13 Radiochemistry Laboratories

The Radiochemistry Laboratories are in the MTR west wing. They are used to support the RML and to perform independent research and development work. Investigators study methods to produce and purify medical radioisotopes and the effects of radiation on hazardous waste.

Laboratories 109 to 112 were used primarily for chemical analysis of reactor primary systems and loop experimental coolants. The predominant radioactivity was from MFPs and MAPs. The south extension to the MTR Wing is the Alpha Laboratories, which was designed for the safe handling of hazardous alpha emitters such as ^{233}U, ^{239}Pu, ^{241}Am, and other TRU materials including ^{252}Cf (Stacy 2000).

Internal exposure potential exists from airborne radioactivity from the large variety of samples [115].

External exposure occurs when irradiated samples are brought into the laboratories. One laboratory has a shielded box, similar to a small hot cell, for handling highly radioactive samples [116].

2.7.14 Liquid Waste Disposal Ponds

The TRA Liquid Waste Ponds are east of the ETR reactor building. The 7.5-acre ponds were built for the disposal of low-level liquid waste from test reactor operations. When all three test reactors were operational, approximately 50 million gal of wastewater per month were discharged to the seepage ponds and the ETR disposal well. Most of the activity pumped to the ponds was ^{51}Cr and ^{3}H (Nebeker and Lakey 1970).

An estimated 3,000 wild ducks per year land on the pond, usually stay less than a week, and have some potential to carry activity off the site. An extensive study analyzed the ducks for ingestion of ^{134}Cs, ^{137}Cs, ^{75}Se, ^{131}I, ^{239}Pu, and ^{240}Pu (Till et al. 2002).

Internal exposure exists from the seepage ponds, which accumulated a significant quantity of MFPs. The activity was fairly stable as long as water levels remained high. When the water was allowed to recede and the soil was allowed to dry, the activity could become airborne by the winds [117].

External exposure occurs to the operators from old accumulated activation products and fission products during routine inspection or sampling of the ponds. Several studies of the water and the soil in the pond resulted in additional external exposure [118].

2.7.15 High-Level Liquid Waste Tanks and Transfer Facility

On the south side of the MTR Health Physics office are four 1,500-gal HLLW catch tanks placed underground in concrete vaults. Tanks 1 and 2 receive water from the Hot Drain System, which includes the MTR floor drains. Tanks 3 and 4 receive liquid waste from the Radiochemistry Laboratory and the TRA Hot Cells. The liquid waste is accumulated until the tank is nearly full and then sampled. If the waste meets low-level waste criteria, it is transferred to the Retention Basin and then to the TRA Liquid Waste Disposal Ponds. If it does not meet the criteria, the waste is transferred to the HLLW holding tanks. When the Hot Waste tanks are nearly full, the wastewater is loaded on a tank truck and shipped to INTEC for processing. The HLLW tanks consist of two 7,500-gal and two 9,000-gal tanks (Stacy 2000).

Internal exposure potential exists from airborne radioactivity during entry to the holding tank pits for repairs from contaminants of MFPs and MAPs [119].

External exposure due to sampling and transferring liquid waste is low because the tanks are in concrete-shielded underground vaults. Radiation sources are MFPs or MAPs when entry to the pits is necessary [120].

2.8 AUXILIARY REACTOR AREA, APRIL 1958 TO LATE 1990s

The ARA was originally the Army Reactor Experimental Area, which changed to the Army Reactor Area. It was established to test stationary, portable, or mobile reactors of low, medium, or high power. The ARA reactors were built and maintained by contractors and a mixed cadre of military personnel trained in the operation of the facilities. ARA is 10 mi east of CFA; it began with the ARA-I site 0.5 mi north of Highway 20. ARA-II, -III, and -IV are at 0.5-mi intervals along an access road, Fillmore

Avenue, north from Highway 20. After the Army phased out its program in about 1965, ARA facilities were used for experiments and tests with multiple radionuclides, particularly at the ARA-I hot cell and laboratory facilities (Stacy 2000). D&D of the ARA ended in 2002.

The following exposures relate to all ARA facilities (Stacy 2000, p.279).

Internal exposure potential was typical of that from operation of a low-power reactor with the exception of the Hot Cell (ARA-I) effluent and the SL-1 excursion, which resulted in fission product release (1,100 Ci to the atmosphere). Airborne radioactivity that consisted of MFPs and MAPs would have caused most internal exposures [121].

External exposure was significant from the SL-1 excursion and recovery operations. Nine technical overexposures resulted that ranged from 15 to 27.3 rem whole-body dose (Horan and Braun 1993, p. 36). Hot cell work and D&D efforts in the reactor areas contributed to exposure pathways. Hot cell entries after work with irradiated experiments were a source of high exposure. Expected exposure potential was from photons greater than 250 keV from the MFPs and MAPs from SL-1, Hot Cell operations, and other work connected with reactor operation and maintenance activities. Dosimetry badges were worn by workers in this area. However, extremity dosimetry might not have been used by all personnel during recovery from the SL-1 accident [122].

2.8.1 ARA-I, Early 1960s to Late 1990s

ARA-I housed a hot cell facility (ARA-25) and laboratory with hoods and metallurgical equipment to support reactor and other radiological experimental work. It operated from the early 1960s through the late 1990s with periods of inactivity. The Hot Cell was involved with recovery of debris from the SL-I excursion accident and associated reactor and fuel experiments. The hot cell and laboratory were involved in a wide spectrum of activities from low-level alpha experiments to work with irradiated reactor samples exposed to core fluxes including melted and destroyed fuel assemblies from reactor tests (Stacy 2000).

2.8.2 ARA-II and SL-1, August 11, 1958 to January 3, 1961

ARA-II, 0.5 mi north of ARA-I, was the site of a low-power BWR that ANL designed and built based on BORAX experience. The reactor was originally named the Argonne Low Power Reactor and was designed to generate only 1 MW. After ANL handed over the finished plant to the Army's operating contractor, Combustion Engineering, the Army named the reactor the Stationary Low-Power Reactor Number 1, or SL-1. SL-1 went critical for the first time on August 11, 1958. It operated for periods of 1 to 6 weeks and then would be shut down for scheduled maintenance and test changes (Stacy 2000, p. 155).

SL-1 was shut down for scheduled annual maintenance on December 23, 1960, and was scheduled for a January 4, 1961, startup. During preparation for the run, the reactor went prompt critical at 9:01 p.m. on January 3, 1961. The criticality created a steam explosion that killed three persons and destroyed the reactor (Stacy 2000, p. 156). The event released fission products (500,000 Ci in the building and 1,100 Ci to the atmosphere; Horan and Braun 1993) and created high-level radioactive contamination to 50 rad/hr around the ARA-II area. Initial recovery from the accident resulted in short-term exposure that exceeded 500 rad/hr to personnel in radiation fields. Extensive cleanup efforts followed and included complete dismantlement of the facility. The reactor vessel went to the TAN, some of the contaminated items went to the RWMC, and some debris was buried in a specially designated location (two large pits and a trench) about 1,600 ft from the SL-1 compound. The walls of the silo, the power conversion and fan-floor equipment, the shielding gravel, and the contaminated

soil that was gathered during the cleanup went into the pits at SL-1. Three feet of clean earth shielded the material. An exclusion fence with hazard warnings around the area remains in place east of the reactor site. Operating power history and release information is in Till et al. (2002, p. 117).

2.8.3 ARA-III, 1960-1961, February 23, 1960, to April 6, 1961

ARA-III, another 0.5 mi farther north, was the site for the Army Gas-Cooled Reactor Experiment (GCRE). GCRE was a water-moderated, nitrogen (gas)-cooled, direct and closed-cycle reactor. It generated 2,200 kWt but no electricity. The Army wanted to develop a mobile nuclear power plant, and the GCRE was the first phase of that program and proved the principle of this concept. The reactor provided engineering and nuclear data for improved components. The GCRE was used to train military and civilian personnel in the operation and maintenance of gas-cooled reactor systems (Stacy 2000, p. 277).

2.8.4 ARA-IV, March 30, 1961, to May 29, 1964

ARA-IV, another 0.5 mi north on Fillmore Avenue, was the site for the Mobile Low-Power Reactor (ML-1). The entire ML-1 plant was designed to be transported either by standard cargo transport planes or standard Army low-bed trailers in separate packages of less than 40 tons each. The ML-1 reactor was operated remotely from a control cab about 500 ft away. It could be moved after a 36-hour shutdown. The reactor was designed for ease of operation and maintenance by technicians at remote installations, for reliable and continuous operation under extreme climatic conditions, and for the rigors of shipment and handling under adverse conditions. The Army phased out its reactor development program in about 1965 (Stacy 2000, p. 279).

2.9 WASTE REDUCTION OPERATIONS COMPLEX, POWER BURST FACILITY, AND SPECIAL POWER EXCURSION REACTOR TEST AREAS

The Waste Reduction Operations Complex, Power Burst Facility (PBF), and SPERT area is south-centrally located east of the CFA on the INL site and 51 mi west of Idaho Falls. The site was originally established to conduct research on small power reactors and reactor safety. Its current mission is storage of spent nuclear fuel, treatment and storage of mixed and low-level wastes, and research to reduce hazardous and mixed wastes. As part of the DOE mandate to treat legacy wastes and remediate the environment, these facilities now provide safe treatment, storage, and recycling of INL radioactive, mixed, and low-level wastes. Many of the treated wastes originated at INL.

The following exposure potentials are for all subsections of Section 2.9.

Internal exposure was possible based on releases from operations at the SPERT reactors and PBF. Maintenance activities and other work with radioactive material (especially from PBF loop experiment) resulted in airborne MFPs and MAPs, which made internal exposure possible; ^{137}Cs was the primary radionuclide [123].

External exposure resulted from experiment changes and maintenance activities. Cesium-137 was a primary nuclide for direct radiation exposure from fission products in the transport lines and in the loops at the PBF during severe fuel damage tests when radiation levels were measured up to 50 rad/hr. Other radiological work activities resulted in much lower exposure rates from the MFPs and MAPs [124].

2.9.1 SPERT-I, June 11, 1955, to 1964

SPERT-I was an open-tank, light-water-moderated, reflected reactor that originally used 92% enriched uranium fuel. The reactor tank was about 4 ft in diameter and 14 ft high and was the first in a series of four safety-testing reactors that were designed to study the behavior of reactors when their power levels changed rapidly. Power runaways were deliberately produced by moving the control rods. The variables in the thousands of SPERT studies included fuel plate design, core configuration, coolant flow, temperature, pressure, reflectors, moderators, and void and temperature coefficients (Stacy 2000, p. 280).

At 12:25 p.m. on November 5, 1962, destructive Test No. 1 was initiated with a plate-type core. A violent explosion occurred immediately after the final power excursion during which complete fuel plate melting occurred in approximately 8% of the core and partial melting occurred in approximately 35%. It was reported that "those isotopes which were collected were released as gases. No solid products were collected." Test No. 2 began at 8:15 a.m. on November 10, 1963, and Test No. 3 began at 1:14 p.m. on April 14, 1964 (Miller, Sola, and McCardell 1964). A number of nondestructive runs were conducted to gain operational information. All operations were conducted from a control building 0.5 mi from the reactor. SPERT-I tests demonstrated the damage-resistant capabilities of low-enrichment (4% ^{235}U) uranium-oxide fuel pins similar to those in water-cooled reactors that power large central stations (Stacy 2000, p. 280).

2.9.2 SPERT-II, March 11, 1960, to October 1964

SPERT-II, south and east of SPERT-I, was an extension of the SPERT-I excursion tests. It was a closed PWR with coolant flow systems that were designed for operation with either light or heavy water. The pressure vessel was 24.5 ft high with a 10-ft inside diameter. Tests with heavy water (which contains deuterium, an isotope of hydrogen) were desired because heavy-water reactors were of growing importance in Canada, Europe, and the United States. In addition, heavy-water tests enabled verification of physics calculations on the effects of neutron lifetime on power excursions (Stacy 2000, p. 280-281).

The area has primarily been used for storage since 1964, and at present is used as a lead storage facility. PBF-contaminated reactor coolant was stored in a tank at the facility and other components were stored in a radioactive material storage area. The radioactive liquid waste and radioactive material storage area have been removed.

2.9.3 SPERT-III, December 19, 1958, to June 1968

SPERT-III was the most versatile facility yet developed for studying the inherent safety characteristics of nuclear reactors. This reactor (which was planned as the third in the series of SPERT reactors but was the second to be built) provided the widest practical range of control over three variables: temperature, pressure, and coolant flow. The reactor was in a pressurized vessel similar to those for commercial power production. Water could flow through the vessel at a rate as high as 20,000 gal/min, and it could sustain temperatures as high as 650°F and pressures as high as 2,500 psi (Stacy 2000, p. 281).

2.9.4 SPERT-IV, July 24, 1962, to August 1970

SPERT-IV was an open-tank, twin-pool facility that permitted detailed studies of reactor stability as affected by variant conditions including forced coolant flow, height of water above the core, hydrostatic head, and other hydrodynamic effects. The water-moderated and reflected reactor used

highly enriched, aluminum-alloyed, plate-type fuel elements. The SPERT-IV facility was modified by the installation of a capsule drive core, which permitted the insertion of fuel samples in a test hole in the center of the reactor core, where they could be subjected to short-period excursions without damaging the fuel in the rest of the core (the "driver" fuel). Work on fuel destructive mechanisms continued until the PBF replaced the capsule drive core operations in SPERT-IV (Stacy 2000, p. 281).

In commercial plants, the reactor cores contain tons of fuel. Analysts imagined the consequences if the coolant somehow failed to carry away the fission heat. Suppose a pipe leaked or broke? The SPERT tests had proven that such a situation would easily put a stop to the chain reaction: the loss of pressure would allow the water to turn to steam, the lower density of steam would fail to moderate the neutrons, and the nuclear reaction would stop. However, the radioactive decay of the fission products in the fuel elements would continue to produce heat and continue to need cooling. This concern spawned the LOCA Program and the PBF. The SPERT reactors were decommissioned and replaced with other operations, as noted below (Stacy 2000, p.281).

2.9.5 Power Burst Facility, September 22, 1972, to 1985

PBF is a much larger and more sophisticated reactor than the SPERT reactors. It was built on the site of the SPERT-I facility. PBF was initially developed to perform tests of nuclear reactor fuels during off-normal reactor operations. It was designed to simulate various kinds of imagined accidents that might be caused by sudden increases in the reactor operating level. PBF was the only reactor in the world that could perform rapid power changes (bursts) within milliseconds. It performed simulated LOCAs and tests of severe fuel rod bursts in a special assembly (loop) in the main reactor core. Fuel damage on experiments in the loop would transport fission products throughout the loop piping and through steam lines outside the shielded loop cubicle. Monitors detected and timed the precise movement of fission products as they escaped from a fuel rod with failed cladding. Data from these tests were used to develop and validate fuel behavior computer programs for the NRC. Retrieval of data and modification of the test configurations resulted in exposure to high radiation fields and potential for release of fission products in the reactor containment (Stacy 2000, p. 280).

The PBF was a high-performance, water-cooled, uranium-oxide-fueled reactor that was designed to provide information on light-water reactors. Airborne effluents were filtered and passed through charcoal beds to remove iodine. Liquid wastes were pumped to a disposal well or held in tanks for transport to INTEC. The reactor operated from September 22, 1972, until 1985 when it was placed on standby status. In 1998, the PBF was placed in shutdown status and is being prepared for fuel removal.

2.9.6 Lead Storage Facility

The Lead Storage Facility is in the old SPERT-II facility. It is used to collect and store clean lead for the INL emergency lead inventory. The building was used in the past for storage of radioactive material. The area is not a radiological concern [125].

2.9.7 Waste Experimental Reduction Facility, 1982 to 2000

The Waste Experimental Reduction Facility is a versatile waste treatment facility that began treating low-level radioactive wastes in 1982 at the location of SPERT-III after that facility's D&D. Its original mission was to reduce the volume of low-level radioactive waste through incineration, stabilization, compaction, and metal sizing processes to prepare the wastes for safe permanent disposition before burial at the RWMC. In the beginning, metal was sized and melted into ingots in two furnaces until it was determined sizing alone was more cost-effective. An incinerator was added and used to reduce

the volume and increase the stability of a wide variety of low-level wastes before disposal at the RWMC. In 1984, the incinerator began treating RCRA-defined mixed waste (radioactive and hazardous). Wastes from INL and other DOE facilities were treated under provisions in the site treatment plan (Stacy 2000).

2.9.8 Mixed Waste Storage Facility

The Mixed Waste Storage Facility in the former SPERT-IV reactor building is a RCRA storage facility for interim storage of mixed low-level wastes. It has regulatory approval to store polychlorinated biphenyls, corrosives, and flammables (Stacy 2000). Treatments are being developed for the types of waste that are stored in the facility.

2.10 ORGANIC MODERATED REACTOR EXPERIMENT, SEPTEMBER 17, 1957, TO APRIL 1963

The Organic-Moderated Reactor Experiment (OMRE) was built a few miles east of the CFA to test the feasibility of the organic-cooled reactor concept. OMRE demonstrated the technical and economic feasibility of using a liquid hydrocarbon as both coolant and moderator. The reactor operated with a succession of cores. The waxy coolant was considered promising because it liquefied at high temperatures but did not corrode metal as water did. In addition, it operated at low pressures, which significantly reduced the risk of leaks. However, it lacked test loops necessary to investigate various organic coolants and experimental fuel elements (Stacy 2000, p. 280).

A scaled-up reactor, the Experimental Organic-Cooled Reactor (EOCR), was built next to OMRE in anticipation of further development of the concept. The purpose of EOCR, which had special testing loops and other advanced features, was to extend and advance the OMRE studies. During the final stages of its construction, EOCR was placed in standby (December 1962) when the AEC decided that the organic-cooled concept would not significantly improve performance over the achievements of other reactor concepts for nuclear power. EOCR never operated. The building was reused for other (non-nuclear) uses (Stacy 2000, p.277).

Internal exposure occurred on November 16, 1960, when an experiment was conducted to determine the feasibility of open-air burning of contaminated solvents that had accumulated at the OMRE facility. Approximately 400 gal of liquid consisting of diesel oil, xylene, methyl-chloroform, and a small amount of water were placed in an open vessel and ignited. Because the reactor did not operate, no other potential for internal dose occurred [126].

External exposure was possible from MFPs and MAPs associated with core changes and associated reactor maintenance in radiological areas [127].

2.11 TEST GRID III, 1957 TO APRIL 24, 1970

Test Grid III was near Lincoln Boulevard south and east of NRF, TRA, and INTEC. It was the site of several tests with atmospheric releases. Operations began in about 1957 and were based on concerns about what would happen from nuclear aircraft crashes and similar events. This highly instrumented grid measured atmospheric conditions for and release information of the tests (Till et al. 2002). These experiments were planned and conducted by the DOE Health Services Laboratory rather than by a contractor.

The following exposure potentials are for all subsections of Section 2.11.

Internal exposure was possible from the airborne radioactivity released from the tests and potential airborne radioactivity from the materials during handling. Grid III was highly instrumented to detect release fractions. FEBT-B had a high ranking for INL releases, and the least amount of monitoring, according to Till et al. (2002). Personnel participated in the site's bioassay program [128].

External exposure would have resulted from working in proximity to and handling and transporting of irradiated test fuel elements with a potential for exposure from photons greater than 250 keV consistent with that of other irradiated fuel loading and unloading tasks [129].

2.11.1 Fuel Element Burn Tests, March 20, 1957

FEBTs A and B were conducted on Grid III to support the General Electric ANP Program to evaluate the consequences of a nuclear aircraft crash with a fire. Aged fuel elements were heated to assist with understanding the behavior of a fuel element in a large fire and to provide initial data on the percentage release of fission products to the environment. Average ground radiation levels near the burn site immediately after the burn test were 200 mrem/hr (Brodsky and Beard 1960). Meteorological conditions had been carefully studied in advance and were closely monitored during the tests.

FEBT-A was conducted at 2:19 p.m. on March 20, 1957, using an irradiated fuel element (well-aged) with 5,000 Ci of fission products. A pool of jet fuel was ignited under the fuel element and reached a temperature of about 2,250˚F. After the fire, the fuel element was intact with a small puncture in the cladding (Stacy 2000, p. 130) [130].

FEBT-B used an induction furnace that heated a fuel element with 10,000 Ci of fission products to 5,000˚F. Most of the fuel element melted and dispersed within 90 seconds during inversion conditions (Brodsky and Beard 1960). The test was conducted at 6:47 p.m. on March 20, 1957. FEBT-B was ranked above other release events at onsite and offsite locations as an episodic event in Till et al. (2002) (Stacy 2000, p. 130).

2.11.2 Fission Products Field Release Tests, July 25, 1958, to September 26, 1958

Fission Products Field Release Tests were conducted on Grid III to represent accidents that involved nuclear-powered aircraft. Nine tests evaluated release percentages, airborne radioactivity, and diffusion and deposition characteristics of fission products that were released from melted aircraft reactor fuel elements (Convair 1959). Five tests were with fuel decayed for 922 to 985 days and four were with fuel decayed for 42 to 65 days. Operating temperatures were between 1,000˚C and 2,300˚C. To simulate a potential accident, the tests used an induction furnace to heat the elements rapidly to the melting point in approximately 2 minutes and maintained this temperature for approximately 10 minutes after melting began (Convair 1959). Instruments situated about a fan-shaped grid with seven concentric arcs and a maximum radius of about 5 mi obtained cloud diffusion, meteorological, radiological, radiobiological, and deposition data. Till et al. (2002) contains information on meteorological conditions, furnace temperatures, release fractions, etc.

2.11.3 Relative Diffusion Tests, November 30, 1967, to October 1, 1969

The four Relative Diffusion Tests involved the intentional release of 1 to 6 Ci of both methyl and elemental radioiodine. Details on these releases are limited, but some information is in DOE (1991b) and Till et al. (2002).

2.11.4 Experimental Cloud Exposure Study, May 3, 1968, to April 24, 1970

The Experimental Cloud Exposure Study tests in 1968 and 1969 consisted of ^{133}Xe releases of 32 to 600 Ci; tests in 1970 consisted of ^{24}Na releases of 6.6 to 120 Ci. The primary objectives for the tests included measuring total exposure at several downwind distances; determining dimensions of the plumes; documenting the release rate and height, wind speed, and temperature; and measuring the gamma energy spectrum at one or more points during the release. Releases were planned to occur during meteorological conditions characterized by winds from the southwest to minimize potential on- and offsite exposure and to ensure that the cloud passed over preset instrumentation. Voillequé (1969) discusses an outline of plans for the ^{133}Xe release tests, including the general objectives and procedures (Till et al. 2002).

2.12 EXPERIMENTAL FIELD STATION AND EXPERIMENTAL DAIRY FARM

The EFS was a 27-acre plot about 7 mi northeast of INTEC near Grid III. This facility, also known as the Dairy Farm, was established to further studies on the pathway of ^{131}I from a release to the human thyroid. It included pastures, a barn, six cows, and a grid of detection instruments in the pasture in regular lines and rows (Stacy 2000).

2.12.1 Controlled Environmental Radioiodine (Release) Tests, May 27, 1963, to December 1977

The primary objective of the Controlled Environmental Radioiodine Test releases was to establish relationships between the amounts of radioiodine in different environmental media. These tests specifically studied relationships between air and soil and vegetation, vegetation and milk, and milk and human thyroids. They involved releases of elemental and methyl radioiodine ranged from 0.05 to 8 Ci. Most of the releases occurred at the Experimental Dairy Farm. Others occurred at INTEC, ARA, NRF, and CFA. Hawley et al. (1964) reports that the release for Controlled Environmental Radioiodine Test No. 1 occurred near ground level over a 30-minute period. In 1968, the name was changed to Controlled Environmental Release Test to reflect the release and study of additional radionuclides such as cesium, cerium, potassium, and krypton. Additional information about this test series is provided by Hawley et al. (1964); Bunch (1966, 1968), and Zimbrick et al. (1969). Early in the test program, the AEC granted permission to seven employees to volunteer to be part of a human experiment program. During initial tests these volunteers sat in the field during the release. In later tests, after the cows had eaten contaminated grass, the volunteers drank small quantities of milk. The series included 29 experiments, but only a few of the early ones involved human consumption of milk [131].

Internal exposures were possible from all the releases and were intentional during the early phases of the tests. Exposures are well documented in the dosimetry records [132].

External exposure was well below the level acceptable for radiological work from the tests. Personnel were required to wear dosimetry devices [133].

2.13 INL RESEARCH CENTER, 1984 TO PRESENT

The INL Research Center was built between 1982 and 1984 on a 35-acre site on North Boulevard in Idaho Falls. The facility was dedicated in 1984 to further the INL research and engineering mission. The INL Research Center has 58 laboratories in IF-603 for geophysics, chemistry, microbiology, and other sciences; 18 of these are general-purpose modules for electronics design, optics, laser and materials testing, and nondestructive examination. The Center conducts laboratory work with tracer-

level radionuclides. One of the early missions was work for the Bureau of Mines with ores, some of which contained natural uranium. The INL Engineering Demonstration Facility (IF-657) houses several prototype-scale research and development projects that support programs in military munitions assay, advanced sensor systems, environmental restoration, subsurface investigation, and materials science. There is a shielded ^{252}Cf source (initially 2.5 mCi) in the high bay of IF-638. The onsite radiological control technician has additional low-energy plutonium, strontium, and americium sources for portable survey instrument response checks [134].

Internal exposure potentials are minimal because of the radiotracer-level, low-energy alpha and beta source materials such as natural uranium, ^{14}C, etc. [135].

External exposure. Low-level personnel exposures have been measured from the neutron source (2- to 20-MeV range) and X-ray (30- to 250-keV range) equipment. Personnel dosimetry is required for all work in radiation areas [136].

2.14 ARMY REENTRY VEHICLE FACILITY SITE (OR STATION), 1965 TO MAY 1996

The Army built the Army Reentry Vehicle Facility Site (ARVFS) 12 mi northeast of CFA in 1965 for classified U.S. Department of Defense experiments with an advanced reentry vehicle fuzing system. The facility consisted of an open-top cylindrical test pit, an underground bunker, and a system of cables and pulleys between the bunker and tank (Thiel 1997; Mobley 1987).

ARVFS was used in 1965 to conduct an irradiation study using four (4) spent MTR fuel elements to evaluate the accuracy of the Radiological Safety Analysis Computer program in predicting cloud-gamma exposure information. Movement of the fuel to conduct the experiment and the transportation of the fuel to and from the facility was an external exposure source. Dose rates are not available (McCaslin 1968).

The bunker was used to store radiologically contaminated NaK coolant from the EBR-I Mark II nuclear reactor core meltdown in November 1955. The NaK was stored from 1974 through 1995, when it was shipped to ANL-W for reprocessing. Engineering evaluations from 1986 through 1992 provided radiological data with maximum radiation exposure rates from the NaK containers to be about 40 R/hr. The reported radiological fission product inventory in the NaK in LaRue and Dolenc (1986) was 23.3 g and about 133 Ci. The Final Safety Analysis Report indicates that the conservative total dose to process the four NaK containers would be 0.394 rem assuming the same person was involved with each step (Mobley and Keller 1991).

Internal exposure potential was minimal at ARVFS because airborne radioactivity was not present during activities at the bunker [137].

External exposure occurred during radiological surveys and loading and unloading of the NaK containers for storage and or transport [138].

2.15 ATTRIBUTIONS AND ANNOTATIONS

Where appropriate in this document, 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 to identify the source and justification for each associated item. Conventional References, which are provided in the next section of this document, link data, quotations, and other information to documents available for review on the Project's Site Research Database (SRDB).

Norman Rohrig served as the initial Document Owner for this document. Mr. Rohrig was previously employed at INL and his work involved management, direction or implementation of radiation protection and/or health physics program policies, procedures or practices related to atomic weapons activities at the site. This revision has been overseen by a new Document Owner, JoAnn Jenkins, who is fully responsible for the content of this document, including all findings and conclusions. In all cases where such information or prior studies or writings are included or relied upon by the Document Owner, those materials are fully attributed to the source.

Donald Marshall served as one of the initial Subject Experts for this document. Mr. Marshall was previously employed at INL and his work involved management, direction or implementation of radiation protection and/or health physics program policies, procedures or practices related to atomic weapons activities at the site. This revision has been overseen by a new Document Owner who is fully responsible for the content, including all findings and conclusions. In all cases where such information or prior studies or writings are included or relied upon by Mr. Marshall, those materials are fully attributed to the source.

Frank Hinckley served as one of the initial Subject Experts for this document. Mr. Hinckley was previously employed at INL and his work involved management, direction or implementation of radiation protection and/or health physics program policies, procedures or practices related to atomic weapons activities at the site. This revision has been overseen by a new Document Owner who is fully responsible for the content, including all findings and conclusions. In all cases where such information or prior studies or writings are included or relied upon by Mr. Hinckley, those materials are fully attributed to the source.

Boyd Leavitt served as one of the initial Subject Experts for this document. Mr. Leavitt was previously employed at INL and his work involved management, direction or implementation of radiation protection and/or health physics program policies, procedures or practices related to atomic weapons activities at the site. This revision has been overseen by a new Document Owner who is fully responsible for the content, including all findings and conclusions. In all cases where such information or prior studies or writings are included or relied upon by Mr. Leavitt, those materials are fully attributed to the source.

Earl Graham served as one of the initial Subject Experts for this document. Mr. Graham was previously employed at ANL-W and his work involved management, direction or implementation of radiation protection and/or health physics program policies, procedures or practices related to atomic weapons activities at the site. This revision has been overseen by a new Document Owner who is fully responsible for the content, including all findings and conclusions. In all cases where such information or prior studies or writings are included or relied upon by Mr. Graham, those materials are fully attributed to the source.

[1] Frank Hinckley. Principal Health Physicist. Intrepid Technology and Resources (ITR). August 2003.
This statement is based on review of ACC (1952) and Cipperley (1958).

[2] Frank Hinckley. Principal Health Physicist. ITR. August 2003.
This statement is based on review of ACC (1952).

[3] Frank Hinckley. Principal Health Physicist. ITR. August 2003.
This statement is based on review of ACC (1952).

[4] Frank Hinckley. Principal Health Physicist. ITR. August 2003.
This statement is based on review of Hoff, Chew, and Rope (1986).

[5] Don Marshall. Principal Health Physicist. ITR. August 2003.
This statement is based on review of Stacy (2000).

[6] Don Marshall. Principal Health Physicist. ITR. August 2003.
The Warm Shop, Hot Shop, and Hot Cells were used for the analysis of various reactor experiments, refueling, and repairs. Reactor fuel contains MFPs due to the fissioning of the uranium and MAPs due to activation from neutron bombardment. It is reasonable to expect that the potential for internal exposure to MFPs and MAPs exists in a facility that handles reactor fuel and components.

[7] Don Marshall. Principal Health Physicist. ITR. August 2003.
This external exposure scenario is routinely included in safety analyses of this type of facility as described in Section 6.3.4 of ORAUT-TKBS-0007-6, *Idaho National Laboratory – Occupational External Dosimetry* (ORAUT 2007a).

[8] Don Marshall. Principal Health Physicist. ITR. August 2003.
This information is based on understanding the safety program and sources of exposure at TAN.

[9] Don Marshall. Principal Health Physicist. ITR. August 2003.
This information is based on understanding the safety program and work experience at TAN.

[10] Don Marshall. Principal Health Physicist. ITR. August 2003.
If a cask leaked, internal exposures could occur because the casks contain spent nuclear fuel under pressure.

[11] Don Marshall. Principal Health Physicist. ITR. August 2003.
External radiation exposures could occur from work in the proximity of the casks due to the elevated radiation levels. Neutron dosimetry is assigned based on the employee's job function and exposure potential; therefore, not all employees are issued neutron dosimetry.

[12] Don Marshall. Principal Health Physicist. ITR. August 2003.
The liquid waste stream contained MFPs and MAPs. External exposures could occur during activities in the vicinity of the liquid waste system due to elevated dose levels.

[13] Don Marshall. Principal Health Physicist. ITR. August 2003.
This information is based on review of Bonney et al. (1995).

[14] Don Marshall. Principal Health Physicist. ITR. August 2003.
This information is based on work experience at the RPSSA. This internal exposure scenario is routinely included in safety analyses of this type of facility as described in Section 5.7.1 of ORAUT-TKBS-0007-5, *Idaho National Laboratory – Occupational Internal Dose* ORAUT (2007b).

[15] Don Marshall. Principal Health Physicist. ITR. August 2003.
This external exposure scenario is routinely included in safety analyses of this type of facility as described in Section 6.3.4 of ORAUT (2007a).

[16] Don Marshall. Principal Health Physicist. ITR. August 2003.
This information is based on work experience at TAN.

[17] Don Marshall. Principal Health Physicist. ITR. August 2003.
This information is based on work experience at TAN.

[18] Don Marshall. Principal Health Physicist. ITR. August 2003.
This external exposure scenario is routinely included in safety analyses of this type of facility as described in Section 6.3.4 of ORAUT (2007a).

[19] Don Marshall. Principal Health Physicist. ITR. August 2003.
This external exposure scenario is routinely included in safety analyses of this type of facility as described in Section 6.3.4 of ORAUT (2007a).

[20] Don Marshall. Principal Health Physicist. ITR. August 2003.
This external exposure scenario is routinely included in safety analyses of this type of facility as described in Section 6.3.4 of ORAUT (2007a).

[21] Don Marshall. Principal Health Physicist. ITR. August 2003.
This information is based on understanding the sources of exposure at TAN and review of Cordes et al. (1965).

[22] Don Marshall. Principal Health Physicist. ITR. August 2003.
The tests were performed with nonradioactive materials.

[23] Don Marshall. Principal Health Physicist. ITR. August 2003.
This information is based on understanding the sources of exposure and work experience at WRRTF.

[24] Don Marshall. Principal Health Physicist. ITR. August 2003.
This information is based on work experience at TAN.

[25] Don Marshall. Principal Health Physicist. ITR. August 2003.
This information is based on understanding the sources of exposure at TAN.

[26] Don Marshall. Principal Health Physicist. ITR. August 2003.
This information is based on understanding the sources of exposure at TAN. This external exposure scenario is routinely included in safety analyses of this type of facility as described in Section 6.3.4 of ORAUT (2007a).

[27] Don Marshall. Principal Health Physicist. ITR. August 2003.
This information is based on review of Kunze and Chase (1970).

[28] Don Marshall. Principal Health Physicist. ITR. August 2003.
This information is based on understanding the sources of exposure at TAN.

[29] Don Marshall. Principal Health Physicist. ITR. August 2003.
This information is based on understanding the sources of exposure at TAN. This external exposure scenario is routinely included in safety analyses of this type of facility as described in Section 6.3.4 of ORAUT (2007a).

[30] Don Marshall. Principal Health Physicist. ITR. August 2003.
This external exposure scenario is routinely included in safety analyses of this type of facility as described in Section 6.3.4 of ORAUT (2007a).

[31] Don Marshall. Principal Health Physicist. ITR. August 2003.
This internal exposure scenario is routinely included in safety analyses of this type of facility as described in Section 5.7.1.1 of ORAUT (2007b).

[32] Don Marshall. Principal Health Physicist. ITR. August 2003.
This information is based on understanding the sources of exposure at TAN. This external exposure scenario is routinely included in safety analyses of this type of facility as described in Section 6.3.4 of ORAUT (2007a).

[33] Boyd Leavitt. Principal Health Physicist. ITR. August 2003.
This internal exposure scenario is routinely included in safety analyses of this type of facility as described in Section 5.7.2 of ORAUT (2007b).

[34] Boyd Leavitt. Principal Health Physicist. ITR. August 2003.
This external exposure scenario is routinely included in safety analyses of this type of facility as described in Section 6.3.4 of ORAUT (2007a).

[35] Boyd Leavitt. Principal Health Physicist. ITR. August 2003.
This information is based on understanding the sources of exposure and work experience at INTEC and on review of Cederberg et al. (1974).

[36] Boyd Leavitt. Principal Health Physicist. ITR. August 2003.
This information is based on review of Stacy (2000).

[37] Boyd Leavitt. Principal Health Physicist. ITR. August 2003.
This information is based on understanding the sources of exposure and work experience at INTEC.

[38] Boyd Leavitt. Principal Health Physicist. ITR. August 2003.
This internal exposure scenario is routinely included in safety analyses of this type of facility as described in Section 5.7.2 of ORAUT (2007b).

[39] Boyd Leavitt. Principal Health Physicist. ITR. August 2003.
This external exposure scenario is routinely included in safety analyses of this type of facility as described in Section 6.3.4 of ORAUT (2007a).

[40] Boyd Leavitt. Principal Health Physicist. ITR. August 2003.
This information is based on understanding the sources of exposure and work experience at INTEC.

[41] Boyd Leavitt. Principal Health Physicist. ITR. August 2003.
This information is based on review of Stroschein and Maeser (1967).

[42] Boyd Leavitt. Principal Health Physicist. ITR. August 2003.
This information is based on understanding the sources of exposure and work experience at INTEC. This external exposure scenario is routinely included in safety analyses of this type of facility as described in Section 6.3.4 of ORAUT (2007a).

[43] Boyd Leavitt. Principal Health Physicist. ITR. August 2003.
This information is based on work experience at the Tank Farm.

[44] Boyd Leavitt. Principal Health Physicist. ITR. August 2003.
This information is based on understanding the sources of exposure and work experience at the Tank Farm. This internal exposure scenario is routinely included in safety analyses of this type of facility as described in Section 5.7.2.2 of ORAUT (2007b).

[45] Boyd Leavitt. Principal Health Physicist. ITR. August 2003.
This information is based on understanding the sources of exposure and work experience at the Tank Farm. This external exposure scenario is routinely included in safety analyses of this type of facility as described in Section 6.3.4 of ORAUT (2007a).

[46] Norm Rohrig. Consulting Health Physicist. ITR. April 25, 2007.
ERDA (1977, p. 106) shows gross beta activity of 4,840 µCi/mL (Ci/m^3) for aluminum-zirconium blend waste and 340 µCi/mL (Ci/m^3) for second- and third-cycle waste in the tank farm.

[47] Boyd Leavitt. Principal Health Physicist. ITR. August 2003.
This information is based on work experience at the WCF.

[48] Boyd Leavitt. Principal Health Physicist. ITR. August 2003.
This information is based on understanding the sources of exposure and work experience at WCF. This internal exposure scenario is routinely included in safety analyses of this type of facility as described in Section 5.7.2.3 of ORAUT (2007b).

[49] Boyd Leavitt. Principal Health Physicist. ITR. August 2003.
This information is based on understanding the sources of exposure and work experience at WCF. This external exposure scenario is routinely included in safety analyses of this type of facility as described in Section 6.3.4 of ORAUT (2007a).

[50] Boyd Leavitt. Principal Health Physicist. ITR. August 2003.
This information is based on understanding the sources of exposure and work experience at FDF.

[51] Boyd Leavitt. Principal Health Physicist. ITR. August 2003.
This information is based on understanding the sources of exposure and work experience at FDF.

[52] Boyd Leavitt. Principal Health Physicist. ITR. August 2003.
This information is based on understanding the sources of exposure and work experience at ICDF.

[53] Boyd Leavitt. Principal Health Physicist. ITR. August 2003.
This information is based on understanding the sources of exposure and work experience at ICDF.

[54] Boyd Leavitt. Principal Health Physicist. ITR. August 2003.
This information is based on understanding the sources of exposure and work experience at ISFSI.

[55] Frank Hinckley. Principal Health Physicist. ITR. August 2003.
This information is based on understanding the sources of exposure and work experience at ISFSI. This external exposure scenario is routinely included in safety analyses of this type of facility as described in Section 6.3.4 of ORAUT (2007a).

[56] Graham, Earl. Principal Health Physicist. ITR. August 2003.
This information is based on review of Haroldsen et al. (1963).

[57] Graham, Earl. Principal Health Physicist. ITR. August 2003.
This information is based on review of Haroldsen et al. (1963).

[58] Graham, Earl. Principal Health Physicist. ITR. August 2003.
This information is based on review of Dietrich, Lichtenberger, and Zinn (1955) and Griffiths, Sill, and Wilhelmsen (1956).

[59] Graham, Earl. Principal Health Physicist. ITR. August 2003.
This information is based on review of Dietrich, Lichtenberger, and Zinn (1955) and Griffiths, Sill, and Wilhelmsen (1956)

[60] Graham, Earl. Principal Health Physicist. ITR. August 2003.
This information is based on review of Dietrich, Lichtenberger, and Zinn (1955) and Griffiths, Sill, and Wilhelmsen (1956)

[61] Graham, Earl. Principal Health Physicist. ITR. August 2003.
This information is based on review of Dietrich, Lichtenberger, and Zinn (1955) and Griffiths, Sill, and Wilhelmsen 1956.

[62] Graham, Earl. Principal Health Physicist. ITR. August 2003.
This information is based on review of Zinn et al. (1956).

[63] Graham, Earl. Principal Health Physicist. ITR. August 2003.
This information is based on review of Zinn et al. (1956).

[64] Graham, Earl. Principal Health Physicist. ITR. August 2003.
This information is based on review of Robertson and Hall (1959).

[65] Graham, Earl. Principal Health Physicist. ITR. August 2003.
This information is based on review of Robertson and Hall (1959).

[66] Graham, Earl. Principal Health Physicist. ITR. August 2003.
This information is based on understanding of the sources of exposure based on experience at BORAX-V.

[67] Graham, Earl. Principal Health Physicist. ITR. August 2003.
This information is based on understanding of the sources of exposure and work experience at BORAX-V.

[68] Graham, Earl. Principal Health Physicist. ITR. August 2003.
This information is based on review of Brittan et al. (1961).

[69] Graham, Earl. Principal Health Physicist. ITR. August 2003.
This information is based on review of Brittan et al. (1961).

[70] Graham, Earl. Principal Health Physicist. ITR. August 2003.
This information is based on understanding of the sources of exposure based on experience at the Argonne Fast Source Reactor and review of Brunson (1959).

[71] Graham, Earl. Principal Health Physicist. ITR. August 2003.
This information is based on understanding of the sources of exposure based on experience at the Argonne Fast Source Reactor and review of Brunson (1959).

[72] Graham, Earl. Principal Health Physicist. ITR. August 2003.
This information is based on review of Freund et al. (1960).

[73] Graham, Earl. Principal Health Physicist. ITR. August 2003.
This information is based on review of Freund et al. (1960).

[74] Graham, Earl. Principal Health Physicist. ITR. August 2003.
This statement is based on understanding of the sources of exposure based on experience at HFEF. Because the hot cells were used to examine irradiated fuel, they would be contaminated with MFPs and MAPs. This contamination is a potential internal exposure hazard during hot cell entries.

[75] Graham, Earl. Principal Health Physicist. ITR. August 2003.
This statement is based on understanding of the sources of exposure based on experience at HFEF. Because the hot cells were used to examine irradiated fuel, they would be contaminated with MFPs and MAPs. This contamination is a potential external exposure hazard during hot cell entries.

[76] Graham, Earl. Principal Health Physicist. ITR. August 2003.
This information is based on understanding of the sources of exposure based on experience at ZPPR and review of Kunze and Chase (1970).

[77] Graham, Earl. Principal Health Physicist. ITR. August 2003.
This information is based on understanding of the sources of exposure based on experience at ZPPR and review of Kunze and Chase (1970).

[78] Graham, Earl. Principal Health Physicist. ITR. August 2003.
This statement by Mr. Graham is based on his understanding of the sources of exposure based on experience at NRAD. Since the hot cells were used to examine irradiated fuel, they would be contaminated with MFPs and MAPs. This contamination is a potential internal exposure hazard.

[79] Graham, Earl. Principal Health Physicist. ITR. August 2003.
This statement by Mr. Graham is based on his understanding of the sources of exposure based on experience at NRAD and review of Richards and McClellan (1979).

[80] Graham, Earl. Principal Health Physicist. ITR. August 2003.
This information is based on understanding of the sources of exposure based on experience at the facility.

[81] Graham, Earl. Principal Health Physicist. ITR. August 2003.
This information is based on understanding of the sources of exposure based on experience at the facility.

[82] Graham, Earl. Principal Health Physicist. ITR. August 2003.
This information is based on understanding of the sources of exposure based on experience at ANL-W.

[83] Graham, Earl. Principal Health Physicist. ITR. August 2003.
This information is based on understanding of the sources of exposure based on experience at ANL-W.

[84] Frank Hinckley. Principal Health Physicist. ITR. August 2003.
This internal exposure scenario is routinely included in safety analyses of this type of facility as described in Section 5.7.4 of ORAUT (2007b).

[85] Frank Hinckley. Principal Health Physicist. ITR. August 2003.
This information is based on understanding the sources of exposure and work experience at RWMC. This external exposure scenario is routinely included in safety analyses of this type of facility as described in Section 6.3.4 of ORAUT (2007a).

[86] Frank Hinckley. Principal Health Physicist. ITR. August 2003.
This information is based on understanding the sources of exposure and work experience at CFA.

[87] Frank Hinckley. Principal Health Physicist. ITR. August 2003.
This information is based on understanding the sources of exposure and work experience at CFA.

[88] Norman Rohrig. Consulting Health Physicist. ITR. August 2003.
This information is based on understanding the sources of exposure and work experience at HPIL. This external exposure scenario is routinely included in safety analyses of this type of facility as described in Section 6.3.4 of ORAUT (2007a).

[89] Norman Rohrig. Consulting Health Physicist. ITR. August 2003.
This information is based on understanding the sources of exposure and work experience at HPIL.

[90] Norman Rohrig. Consulting Health Physicist. ITR. August 2003.
This information is based on understanding the sources of exposure and work experience at HPIL.

[91] Norman Rohrig. Consulting Health Physicist. ITR. August 2003.
The AmBe source was transferred to Idaho State University. This information is based on the experience of Mr. Rohrig at INL.

[92] Frank Hinckley. Principal Health Physicist. ITR. August 2003.
This information is based on understanding the sources of exposure and work experience at CFA.

[93] Don Marshall. Principal Health Physicist. ITR. August 2003.
This information is based on work experience at the MTR area.

[94] Don Marshall. Principal Health Physicist. ITR. August 2003.
This internal exposure scenario is routinely included in safety analyses of this type of facility as described in Section 5.7.6 of ORAUT (2007b).

[95] Don Marshall. Principal Health Physicist. ITR. August 2003.
This external exposure scenario is routinely included in safety analyses of this type of facility as described in Section 6.3.4 of ORAUT (2007a).

[96] Don Marshall. Principal Health Physicist. ITR. August 2003.
This internal exposure scenario is routinely included in safety analyses of this type of facility as described in Section 5.7.6 of ORAUT (2007b).

[97] Don Marshall. Principal Health Physicist. ITR. August 2003.
This information is based on understanding the sources of exposure and work experience at ETR. This external exposure scenario is routinely included in safety analyses of this type of facility as described in Section 6.3.4 of ORAUT (2007a).

[98] Don Marshall. Principal Health Physicist. ITR. August 2003.
This internal exposure scenario is routinely included in safety analyses of this type of facility as described in Section 5.7.6 of ORAUT (2007b).

[99] Don Marshall. Principal Health Physicist. ITR. August 2003.
This information is based on understanding the sources of exposure and work experience at ATR. This external exposure scenario is routinely included in safety analyses of this type of facility as described in Section 6.3.4 of ORAUT (2007a).

[100] Don Marshall. Principal Health Physicist. ITR. August 2003.
This internal exposure scenario is routinely included in safety analyses of this type of facility as described in Section 5.7.6 of ORAUT (2007b).

[101] Don Marshall. Principal Health Physicist. ITR. August 2003.
This information is based on understanding the sources of exposure and work experience at MTR. This external exposure scenario is routinely included in safety analyses of this type of facility as described in Section 6.3.4 of ORAUT (2007a).

[102] Don Marshall. Principal Health Physicist. ITR. August 2003.
This internal exposure scenario is routinely included in safety analyses of this type of facility as described in Section 5.7.6 of ORAUT (2007b).

[103] Don Marshall. Principal Health Physicist. ITR. August 2003.
This information is based on understanding the sources of exposure and work experience at ARMF. This external exposure scenario is routinely included in safety analyses of this type of facility as described in Section 6.3.4 of ORAUT (2007a).

[104] Don Marshall. Principal Health Physicist. ITR. August 2003.
This internal exposure scenario is routinely included in safety analyses of this type of facility as described in Section 5.7.6 of ORAUT (2007b).

[105] Don Marshall. Principal Health Physicist. ITR. August 2003.
This information is based on understanding the sources of exposure and on work experience at the facility. This external exposure scenario is routinely included in safety analyses of this type of facility as described in Section 6.3.4 of ORAUT (2007a).

[106] Don Marshall. Principal Health Physicist. ITR. August 2003.
This information is based on understanding the sources of exposure and work experience at the ETR Critical Facility. This internal exposure scenario is routinely included in safety analyses of this type of facility as described in Section 5.7.6 of ORAUT (2007b).

[107] Don Marshall. Principal Health Physicist. ITR. August 2003.
This information is based on understanding the sources of exposure and work experience at the ETR Critical Facility. This external exposure scenario is routinely included in safety analyses of this type of facility as described in Section 6.3.4 of ORAUT (2007a).

[108] Don Marshall. Principal Health Physicist. ITR. August 2003.
This information is based on understanding the sources of exposure and work experience at the ATR Critical Facility. This internal exposure scenario is routinely included in safety analyses of this type of facility as described in Section 5.7.6 of ORAUT (2007b).

[109] Don Marshall. Principal Health Physicist. ITR. August 2003.
This internal exposure scenario is routinely included in safety analyses of this type of facility as described in Section 5.7.6 of ORAUT (2007b) and is supported by Stroschein and Maeser (1967).

[110] Don Marshall. Principal Health Physicist. ITR. August 2003.
This external exposure scenario is routinely included in safety analyses of this type of facility as described in Section 6.3.4 of ORAUT (2007a) and is supported by Stroschein and Maeser (1967).

[111] Don Marshall. Principal Health Physicist. ITR. August 2003.
This internal exposure scenario is routinely included in safety analyses of this type of facility as described in Section 5.7.6 of ORAUT (2007b).

[112] Don Marshall. Principal Health Physicist. ITR. August 2003.
This external exposure scenario is routinely included in safety analyses of this type of facility as described in Section 6.3.4 of ORAUT (2007a).

[113] Don Marshall. Principal Health Physicist. ITR. August 2003.
This internal exposure scenario is routinely included in safety analyses of this type of facility as described in Section 5.7.6 of ORAUT (2007b).

[114] Don Marshall. Principal Health Physicist. ITR. August 2003.
This facility is a low background facility and highly radioactive sources/samples are not allowed in.

[115] Don Marshall. Principal Health Physicist. ITR. August 2003.
This internal exposure scenario is routinely included in safety analyses of this type of facility as described in Section 5.7.6 of ORAUT (2007b).

[116] Don Marshall. Principal Health Physicist. ITR. August 2003.
This external exposure scenario is routinely included in safety analyses of this type of facility as described in Section 6.3.4 of ORAUT (2007a).

[117] Don Marshall. Principal Health Physicist. ITR. August 2003.
This information is based on review of Nebeker and Lakey (1970), Till et al. (2002), and ERDA (1977).

[118] Don Marshall. Principal Health Physicist. ITR. August 2003.
This information is based on review of Nebeker and Lakey (1970), Till et al. (2002), and ERDA (1977).

[119] Don Marshall. Principal Health Physicist. ITR. August 2003.
This information is based on review of Nebeker and Lakey (1970) and ERDA (1977).

[120] Don Marshall. Principal Health Physicist. ITR. August 2003.
This information is based on review of Nebeker and Lakey (1970) and ERDA (1977).

[121] Frank Hinckley. Principal Health Physicist. ITR. August 2003.
This information is based on review of Horan and Gammil (1963).

[122] Frank Hinckley. Principal Health Physicist. ITR. August 2003.
This information is based on review of Horan and Gammil (1963).

[123] Frank Hinckley. Principal Health Physicist. ITR. August 2003.
This information is based on review of INEEL (2001) and ERDA (1977).

[124] Frank Hinckley. Principal Health Physicist. ITR. August 2003.
This information is based on review of ERDA (1977).

[125] Frank Hinckley. Principal Health Physicist. ITR. August 2003.
This storage area is not a radiological concern since the lead stored here is of the “low background” variety.

[126] Frank Hinckley. Principal Health Physicist. ITR. August 2003.
This information is based on review of ERDA (1977) and Rielly (1998).

[127] Frank Hinckley. Principal Health Physicist. ITR. August 2003.
This external exposure scenario is routinely included in safety analyses of this type of facility as described in Section 6.3.4 of ORAUT (2007a).

[128] Frank Hinckley. Principal Health Physicist. ITR. August 2003.
This information is based on review of SCA, SRA, and WCC (1994) and DOE (1991a).

[129] Frank Hinckley. Principal Health Physicist. ITR. August 2003.
This information is based on review of SCA, SRA, and WCC (1994) and DOE (1991a).

[130] Frank Hinckley. Principal Health Physicist. ITR. August 2003.
This information is based on review of SCA, SRA, and WCC (1994) and DOE (1991a).

[131] Frank Hinckley. Principal Health Physicist. ITR. August 2003.
This information is based on review of Bunch (1968), Hawley et al. (1964), and Zimbrick et al. (1969).

[132] Frank Hinckley. Principal Health Physicist. ITR. August 2003.
This information is based on review of Bunch (1968), Hawley et al. (1964), and Zimbrick et al. (1969).

[133] Frank Hinckley. Principal Health Physicist. ITR. August 2003.
This information is based on review of Bunch (1968), Hawley et al. (1964), and Zimbrick et al. (1969).

[134] Frank Hinckley. Principal Health Physicist. ITR. August 2003.
This information is based on review of Rielly (1998).

[135] Frank Hinckley. Principal Health Physicist. ITR. August 2003.
This information is based on review of Rielly (1998).

[136] Frank Hinckley. Principal Health Physicist. ITR. August 2003.
This information is based on review of Rielly (1998).

[137] Frank Hinckley. Principal Health Physicist. ITR. August 2003.
This information is based on review of Mobley and Keller (1991), Thiel (1997), and Mobley (1987).

[138] Frank Hinckley. Principal Health Physicist. ITR. August 2003.
This information is based on review of Mobley and Keller (1991), Thiel (1997), and Mobley (1987).

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Cordes, O. L., R. P. Bird, G. A. Dineen, and J. R. Fielding, 1965, *Radiological Aspects of SNAPTRAN 2/10A-3 Destructive Test*, IDO-17038, Phillips Petroleum Company, National Reactor Testing Station, Idaho Falls, Idaho, January. [SRDB Ref ID: 1122]

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GLOSSARY

activation
Creation of a radioisotope by interaction of a stable (nonradioactive) element with neutrons, protons, or other types of radiation.

americium–beryllium (AmBe)
Common neutron source created by an alpha particle from ^{241}Am interacting with beryllium to produce a large neutron yield with low gamma-ray yield.

background radiation
Radiation from cosmic sources, naturally occurring radioactive materials including naturally occurring radon, and global fallout from the testing of nuclear explosives. Background radiation does not include radiation from source, byproduct, or Special Nuclear Materials regulated by the U.S. Nuclear Regulatory Commission. The average individual exposure from background radiation is about 360 millirem per year.

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

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

blowdown
Sudden depressurization from a break in a pipe containing pressurized water in a reactor system.

boiling-water reactor (BWR)
Nuclear reactor in which boiling water in the core serves as both coolant and moderator.

breeder reactor
Nuclear reactor in which the operation produces a net increase in fissionable reactor fuel.

calcine
(1) Dry solid (grainy or granular) product of a chemical process that removes liquids from a solution. (2) Process for creating the chemical reaction that removes liquids from a solution.

cladding
Outer layer of metal that encases a reactor fuel element or fissile material of the pit of a nuclear weapon, often made with aluminum or zirconium. In a reactor, cladding promotes the transfer of heat from the fuel to the coolant, and it builds up fission and activation products over time from the fission of the fuel.

containment

(1) State of being contained. (2) Heavy structure completely surrounding a source of radioactivity to prevent it from escaping or affecting the environment.

contamination

Radioactive material in an undesired location including air, soil, buildings, animals, and persons.

control rod

Neutron-absorbing device in a reactor used to slow or speed the reaction. Also called safety rod.

control room

Room in a nuclear reactor facility that houses the controls for operation and monitoring.

coolant

Gas or liquid in a nuclear reactor that removes the heat from the fission process.

core

Central region of a nuclear reactor where fission of the fuel takes place.

criticality

State of a radioactive mass (e.g. the core of a nuclear reactor) when the fission reaction becomes self-sustaining. Nuclear reactors *go critical* when started.

curie (Ci)

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

decommissioning

Removal of a facility from service, usually involving decontamination of radioactivity to specified levels and often involving demolition of the facility.

decontamination

Reduction or removal of radioactive material from a structure, area, object, or person. Decontamination can occur through (1) treating the surface to remove or decrease the contamination or (2) allowing natural radioactive decay to occur over time.

depleted uranium (DU)

Uranium with a percentage of ^{235}U lower than the 0.7% found in natural uranium. As examples, spent (used) fuel elements, byproduct tails, residues from uranium isotope separation, and some weapons materials contain DU. DU can be blended with HEU to make reactor fuel or used as a raw material to produce plutonium.

dose

In general, the effects of ionizing radiation in terms of the specific amount of energy absorbed per unit of mass. Effective and equivalent doses are in units of rem or sievert; other types of dose are in units of roentgens, rads, reps, or grays.

dosimeter

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

dosimetry

Measurement and calculation of internal and external radiation doses.

DOT-6M

Drums that hold DOT-2R containers and meet the specifications of 49 C.F.R 178.354.

emergency core cooling system

Backup system that injects coolant into the core of a reactor to prevent overheating of the fuel and subsequent fuel damage. See *meltdown*.

enriched uranium

Uranium in which processing has increased the proportion of ^{235}U to ^{238}U to above the natural level of 0.7%. Reactor-grade uranium is usually about 3.5% ^{235}U; weapons-grade uranium contains greater than 90% ^{235}U.

excursion

Planned or accidental increase in the normal operating power level of nuclear reactions.

exposure

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

extremities

The portion of the arm from and including the elbow through the fingertips and the portion of the leg from and including the knee and patella through the toes.

film

Radiation-sensitive photographic film in a light-tight wrapping.

fission

Splitting of the nucleus of an atom (usually of a heavy element) into at least two other nuclei and the release of a relatively large amount of energy. This transformation usually releases two or three neutrons.

fission product

(1) Radionuclides produced by fission or by the subsequent radioactive decay of radionuclides. (2) Fragments other than neutrons that result from the splitting of an atomic nucleus.

flux

Rate of flow of mass, volume, or energy. Flux is often used to mean *flux density*.

flux density

Flux per unit of cross-sectional area perpendicular to the direction of flow. In relation to health physics, usually describes the flow of neutrons or photons in a radiation field or X-ray beam.

fuel assembly

Arrangement of nuclear fuel for use in a nuclear reactor. Fuel is most commonly in the form of pellets arranged in rods of various diameters but can also be formed into plates or other shapes.

gamma radiation

Electromagnetic radiation (photons) of short wavelength and high energy (10 kiloelectron-volts to 9 megaelectron-volts) that originates in atomic nuclei and accompanies many nuclear reactions (e.g., fission, radioactive decay, and neutron capture). Gamma rays are very penetrating, but dense materials such as lead or uranium or thick structures can stop them. Gamma photons are identical to X-ray photons of high energy; the difference is that X-rays do not originate in the nucleus.

half-life

Time in which half of a given quantity of a particular radionuclide disintegrates (decays) into another nuclear form. During one half-life, the number of atoms of a particular radionuclide decreases by one half. Each radionuclide has a unique half-life ranging from millionths of a second to billions of years.

highly enriched uranium

Uranium enriched to at least 20% ^{235}U for use as fissile material in nuclear weapons components and some reactor fuels. Also called high-enriched uranium.

hot cell

Specialized shielded laboratory in which radioactive materials may be handled with the aid of remotely operated manipulators. The walls and windows of the laboratory are made of materials designed to protect workers from radiation.

hot run

Operational or test run of a chemical process and equipment using radioactive materials.

interim storage

Storage between use and final disposition.

ionizing radiation

Radiation of high enough energy to remove an electron from a struck atom and leave behind a positively charged ion. High enough doses of ionizing radiation can cause cellular damage. Ionizing particles include alpha particles, beta particles, gamma rays, X-rays, neutrons, high-speed electrons, high-speed protons, photoelectrons, Compton electrons, positron/negatron pairs from photon radiation, and scattered nuclei from fast neutrons. See *beta radiation*, *gamma radiation*, *neutron radiation*, *photon radiation*, and *X-ray radiation*.

irradiate

To expose to ionizing radiation.

isotope

One of two or more atoms of a particular element that have the same number of protons (atomic number) but different numbers of neutrons in their nuclei (e.g., ^{234}U, ^{235}U, and ^{238}U). Isotopes have very nearly the same chemical properties but often have different physical properties.

mega-

Prefix that multiplies a unit by 1 million (1×10^{6}).

meltdown

Melting of nuclear reactor fuel caused by a failure of the coolant to adequately carry away heat.

micro-

Prefix that divides a unit by 1 million (multiplies by 1×10^{-6}).

milli-

Prefix that divides a unit by 1,000 (multiplies by 1×10^{-3}).

mixed waste

Unwanted material containing both radioactive and hazardous components.

moderator

Material such as water, heavy water, or graphite that slows the neutrons in a reactor to cause more fission.

natural uranium

Uranium as found in nature, approximately 99.27% ^{238}U, 0.72% ^{235}U, and 0.0054% ^{234}U by weight. The specific activity of this mixture is 2.6×10^{7} becquerel per kilogram (0.7 picocuries per gram). See *uranium*.

neutron

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

neutron, fast

Neutron with energy equal to or greater than 10 keV.

neutron, thermal

Strictly, neutron in thermal equilibrium with surroundings. In general, neutrons with energy less than about 0.5 eV.

neutron radiation

Radiation that consists of free neutrons unattached to other subatomic particles emitted from a decaying radionuclide. Neutron radiation can cause further fission in fissionable material such as the chain reactions in nuclear reactors, and nonradioactive nuclides can become radioactive by absorbing free neutrons. See *neutron*.

nuclear energy

Energy released by nuclear reaction, some of which can be ionizing radiation. Of particular importance is the energy released when a neutron initiates fission or when two nuclei join together under millions of degrees of heat (fusion). Also called atomic energy.

nuclear power plant

Electrical generating facility using nuclear fuel.

nucleus

Central core of an atom, which consists of positively charged protons and, with the exception of ordinary hydrogen, electrically neutral neutrons. The number of protons (atomic number) uniquely defines a chemical element, and the number of protons and neutrons is the mass number of a nuclide. The plural is nuclei.

nuclide

Stable or unstable isotope of any element. Nuclide relates to the atomic mass, which is the sum of the number of protons and neutrons in the nucleus of an atom. A radionuclide is an unstable nuclide.

photon

Basic unit of electromagnetic radiation. Photons are massless "packages" of light energy that range from low-energy microwave photons to high-energy gamma rays. Photons have energies between 10 and 100 kiloelectron-volts. See *photon radiation*.

photon radiation

Electromagnetic radiation of light energy (photons) from microwaves to gamma rays. Gamma rays and X-rays are examples of ionizing photon radiation, which have enough energy to penetrate matter, including the body, and deposit energy in that matter.

pocket ionization chamber

Cylindrical monitoring device commonly clipped to the shirt or laboratory coat pocket to measure ionizing radiation. Also called pencil, pocket pencil, pencil dosimeter, and pocket dosimeter.

pressurized-water reactor (PWR)

Nuclear reactor in which water is kept under pressure in a vessel to prevent boiling.

proton

Basic nucleic particle with a positive electrical charge and mass slightly less than that of a neutron. There are protons in the nuclei of every atom, and the number of protons is the atomic number, which determines the chemical element.

rad

Traditional unit for expressing absorbed radiation dose, which is the amount of energy from any type of ionizing radiation deposited in any medium. A dose of 1 rad is equivalent to the absorption of 100 ergs per gram (0.01 joule per kilogram) of absorbing tissue. The rad has been replaced by the gray in the International System of Units (100 rads = 1 gray). The word derives from radiation absorbed dose.

radiation

Subatomic particles and electromagnetic rays (photons) that travel from one point to another, some of which can pass through or partly through solid materials including the human body. See *ionizing radiation*.

radioactive

Giving off ionizing radiation such as alpha particles or X-rays.

radioactivity

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

radioactive waste

Radioactive solid, liquid, and gaseous materials for which there is no further use. Wastes are generally classified as high-level (with radioactivity as high as hundreds of thousands of curies per gallon or cubic foot), low-level (in the range of 1 microcurie per gallon or cubic foot), intermediate level (between these extremes), mixed (also contains hazardous waste), and transuranic.

radionuclide

Radioactive nuclide. See *radioactive* and *nuclide*.

radioactive lanthanum (RaLa)

Isotope ^{140}La, which was tested for application in nuclear weapons.

reactor

Device in which a fission chain reaction occurs under controlled conditions to produce heat or useful radiation for experimental purposes or to generate electrical power or nuclear fuel.

reactor vessel

Structure enclosing the fuel elements, control elements, coolant piping, and other structures that support the reactor core.

reflector

Part of the structure of some nuclear reactors that reflects neutrons back into the reactor core.

rem

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

retention basin

Outdoor basin (any of several designs) in which liquid solutions are deposited and held pending evaporation or the precipitation of solids.

reprocessing

Mechanical and chemical processing of spent nuclear fuel to separate useable fissionable products (i.e., uranium and plutonium) from waste material. Reprocessing was discontinued in the United States in 1992.

roentgen (R)

Unit of photon (gamma or X-ray) exposure for which the resultant ionization liberates a positive or negative charge equal to 2.58×10^{-4} coulombs per kilogram (or 1 electrostatic unit of electricity per cubic centimeter) of dry air at 0°C and standard atmospheric pressure. An exposure of 1 R is approximately equivalent to an absorbed dose of 1 rad in soft tissue for higher energy photons (generally greater than 100 kiloelectron-volts).

Semiscale

Informal name of a scale model of a nuclear reactor operated at INL. The core simulated the heat of a nuclear reaction by electrical means and was used to study the behavior of water and steam in accidents involving the loss of coolant caused by a break in a coolant system.

shielding

Material or obstruction that absorbs ionizing radiation and tends to protect personnel or materials from its effects.

spent nuclear fuel

Reactor fuel containing fission and activation products that can no longer economically sustain a chain reaction.

thermoluminescent dosimeter

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

transuranic (TRU) elements

Elements with atomic numbers above 92 (uranium). Examples include plutonium and americium. All isotopes of the transuranic elements are radioactive, they are naturally either rare or nonexistent on Earth, and most are known only as a result of research using nuclear reactors and particle accelerators because of extremely short half-lives.

transuranic (TRU) waste

Radioactive waste that contains transuranic elements and has radioactivity of 100 or more nanocuries per gram.

tritium

Radioactive isotope of hydrogen that contains one proton and two neutrons in its nucleus. Because tritium is chemically identical to the natural hydrogen atoms in water, it can be easily ingested. It decays by beta emission and has a radioactive half-life of about 12.5 years. Its uses include increasing the yield of a nuclear weapon.

U.S. Atomic Energy Commission (AEC)

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

the AEC in 1974. The U.S. Department of Energy succeeded the U.S. Energy Research and Development Administration in 1979.

uranium (U)

Heavy, metallic, and radioactive element with atomic number 92. Most natural uranium as found in ores is ^{238}U with trace levels of other isotopes. Uranium-235 (0.7% of natural uranium) is fissile by itself and used in nuclear weapons as well as reactors. Uranium-238 (99.3% of natural uranium) is fissionable by fast neutrons and used in nuclear reactors. Natural uranium contains a minute amount of ^{234}U. See *depleted uranium*, *enriched uranium*, *highly enriched uranium*, and *natural uranium*.

uranium oxide

Metallic compound of uranium and oxygen useful as nuclear fuel because it has a higher melting point than metallic uranium. However, its heat transfer properties are not as efficient as those of metallic uranium.

waste storage tank

Holding tank for liquid or gaseous wastes that might or might not be radioactive.

water-moderated reactor

Reactor in which water slows the speed of neutrons from fissioning atoms to increase the number of neutrons that cause fission.

whole-body dose

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

X-ray radiation

Penetrating electromagnetic radiation (photons) of short wavelength (0.001 to 10 nanometers) and energy less than 250 kiloelectron-volts. X-rays usually come from excitation of the electron field around certain nuclei. Once formed, there is no difference between X-rays and gamma rays, but gamma photons originate inside the nucleus of an atom.

zero power

Reactor power level that maintains a chain reaction at an extremely low power level producing very little heat. Also called low power.

zirconium

Metallic element with atomic number 40. Zirconium is highly resistant to corrosion, and it is alloyed with aluminum to make cladding for nuclear fuel and sometimes in small amounts with the fuel itself.