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ACRONYMS AND ABBREVIATIONS

AEC   U.S. Atomic Energy Commission
AFSR   Argonne Fast Source Reactor
ANL   Argonne National Laboratory
ANL-E   Argonne National Laboratory–East
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)

BNFL   British Nuclear Fuels Limited
BORAX   Boiling-Water Reactor Experiment
BWR   boiling-water reactor

CAM   continuous air monitor
CDC   capsule drive core
CE   Critical Experiment
CERCLA   Comprehensive Environmental Response, Compensation, and Liability Act
CERT   Controlled Environmental Radioiodine (later Release) Test
CET   Critical Experiment Tank
CFA   Central Facilities Area
CFRMF   Coupled Fast Reactivity Measurement Facility
Ci   curie
CPP   Chemical Processing Plant
CRCE   Cavity Reactor Critical Experiment
CTF   Core Test Facility

d   day
D&D   decontamination and decommissioning
DOD   U.S. Department of Defense
DOE   U.S. Department of Energy
DOELAP   DOE Laboratory Accreditation Program
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)

FASB   Fuel Assembly and Storage Building
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
ft   foot

g   gram
gal   gallon
GCRE   Gas-Cooled Reactor Experiment
GE   General Electric Company

HEU   highly enriched uranium
HFEF   Hot Fuel Examination Facility
HLLW   high-level liquid waste
HOTCE   Hot Critical Experiment
HP   health physics
HPIL   Health Physics Instrument Laboratory
HPP   Hot Pilot Plant (later Headend Processing Plant)
hr   hour
HTRE   Heat Transfer Reactor Experiment

IC   Initial Criticality
ICDF   INL CERCLA Disposal Facility
ICP   Idaho Cleanup Project
ICPP   Idaho Chemical Processing Plant
ID, IDO   Idaho Operations Office
IET   Initial Engine Test
IFSF   Irradiated Fuel Storage Facility
IMBA   Integrated Modules for Bioassay Analysis
in.   inch
INEEL   Idaho National Engineering and Environmental Laboratory
INL   Idaho National Laboratory
INTEC   Idaho Nuclear Technology and Engineering Complex
IRC   INL Research Center
IREP   Interactive RadioEpidemiological Program
ISFSI   independent spent fuel storage installation
IWP   Industrial Waste Pond

keV   kiloelectron-volt, 1,000 electron-volts
kg   kilogram
kVp   kilovolts-peak, applied kilovoltage
kW   kilowatt

L   liter
LOCA   loss-of-coolant accident
LOFT   Loss-of-Fluid Test
LPTF   Low Power Test Facility

m   meter
MAP   mixed activation product
mCi   millicurie
MeV   megaelectron-volt, 1 million electron-volts
MFP   mixed fission product

min   minute
mL   milliliter
ML-1   Mobile Low-Power Reactor No. 1
mo   month
mph   miles per hour
mR   milliroentgen
mrem   millirem
ms   millisecond
MTR   Materials Test Reactor
MW   megawatt

NASA   National Aeronautics and Space Administration
nCi   nanocurie
NIOSH   National Institute for Occupational Safety and Health
NRAD   Neutron Radiography Facility
NRC   U.S. Nuclear Regulatory Commission
NRF   Naval Reactors Facility
NRTS   National Reactor Testing Station
NWCF   New Waste Calcining Facility

OMRE   Organic-Moderated Reactor Experiment
ORAU   Oak Ridge Associated Universities

PBF   Power Burst Facility
PIC   pocket ionization chamber
PIF   Process Improvement Facility
POC   probability of causation
psi   pounds per square inch
psig   pounds per square inch gauge
PWR   pressurized-water reactor

R   roentgen
R&D   research and development
RAL   Remote Analytical Laboratory
RaLa   radioactive lanthanum
RAM   remote area monitor
RCRA   Resource Recovery and Conservation Act
RESL   Radiological and Environmental Sciences Laboratory
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
SCRCE   Spherical Cavity Reactor Critical Experiment
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)
SRS   Savannah River Site
STP   Sewage Treatment Pond
STPF   Shield Test Pool Facility
STR   Submarine Thermal Reactor (S1W)
SWEPP   Stored Waste Examination Pilot Plant

t   thermal
TAN   Test Area North
TBD   technical basis document
TLD   thermoluminescent dosimeter
TMI   Three Mile Island
TRA   Test Reactor Area
TREAT   Transient Reactor Experiment and Test (facility)
TRIGA   Training, Research, Isotope General Atomics (reactor)
TSA   Transuranic Storage Area
TRU   transuranic
TSF   Technical Support Facility

U.S.C.   United States Code

W   watt
WCF   Waste Calcining Facility
WERF   Waste Experimental Reduction Facility
WIPP   Waste Isolation Pilot Plant
wk   week
WRRTF   Water Reactor Research Test Facility

yd   yard

ZPR   Zero Power Reactor
ZPPR   Zero Power Plutonium (later Physics) Reactor

α   alpha particle

β   beta particle

γ   gamma

μCi   microcurie

°C   degrees Celsius
°F   degrees Fahrenheit

§   section or sections

2.1   INTRODUCTION

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

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

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

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

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

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

2.1.1 **Purpose**

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

INL played a major role in early reactor research and development. The site has operated 52 reactors plus fuel handling and reprocessing and radioactive waste storage and disposal facilities since it began operations in 1949. The purpose of this TBD is to assist in the evaluation of worker dose from INL processes using the methodology in the *External Dose Reconstruction Implementation Guideline* (NIOSH 2006) and the *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 occupational dose that can be reasonably associated with worker radiation exposure covered under the EEOICPA legislation. The TBD addresses evaluation of monitored and unmonitored worker exposure and missed dose. In addition, it presents the technical basis of methods used to prepare INL worker dose records for input to the NIOSH Interactive RadioEpidemiological Program (IREP) and the Integrated Modules for Bioassay Analysis (IMBA) computer codes used to evaluate worker dose. It also presents information on the uncertainty for recorded INL exposure and dose values.

This section describes the facilities and processes and historical information related 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.

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, situated on the Snake River Plain of southeastern Idaho at an elevation of about 5,000 ft, is above the Snake River Plain Aquifer.

In 1949, the U.S. Atomic Energy Commission (AEC) established the National Reactor Testing Station (NRTS) in Idaho as a Federal reservation to build, test, and operate nuclear reactors. The site utilized a variety of support facilities and equipment. In 1974, NRTS became the Idaho National Engineering Laboratory, which in 1997 became the Idaho National Engineering and Environmental Laboratory (INEEL). On February 1, 2005, the site became the Idaho National Laboratory, combining the research side of the INEEL and Argonne National Laboratory–West (ANL-W) and the Idaho Cleanup Project (ICP) working on closure of inactive portions of the site. (For convenience, this TBD refers to the site as INL.)

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 historical prime site contractors and the years during which they operated. 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).

Operating years	Prime contractor
1950–1966	Phillips Petroleum Company
1966–1972	Idaho Nuclear Corporation (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, LLC
2005	Battelle Energy Alliance for INL, CH2M Washington Group for ICP

Table 2-2. Other operating facility technical area contractors (Stacey 2000).

Argonne National Laboratory–West	
1949–2004	University of Chicago
Idaho Chemical Processing Plant (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 INEL
1991–1994	Babcock & Wilcox Idaho, Inc.

Dosimetry services at INL were unique among DOE facilities in that DOE Idaho Operations Office (ID) personnel operated and provided internal and external dosimetry services. The DOE-ID dosimetry branch provided and analyzed external dosimetry badges, counted workers in the whole-body counter, and analyzed bioassay samples. DOE-ID also provided portable radiation survey instruments and maintained and calibrated them. Originally, DOE-ID personnel at the Radiological and Environmental Sciences Laboratory (RESL) 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), dosimetry responsibility transferred to the prime contractor on January 2, 1989, to eliminate a conflict of interest on the part of DOE-ID.

Figure 2-1 shows the relative location of each facility or technical area 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 exempted under EEOICPA. The subsections for the facilities and technical areas and processes contain information on the particular area.

Personnel working at INL in designated radiological areas were typically required to wear state-of-the-art dosimetry [film badges, thermoluminescent dosimeters (TLDs), pocket ion chambers (PICs), respiratory protection, anticontamination clothing, etc.] [1]. Facilities and radiological areas were monitored by remote and portable remote area monitors (RAMs) and continuous air monitors (CAMs). Portable RAMs, CAMs, air samplers, etc., were used for work where fixed units were not available [2].
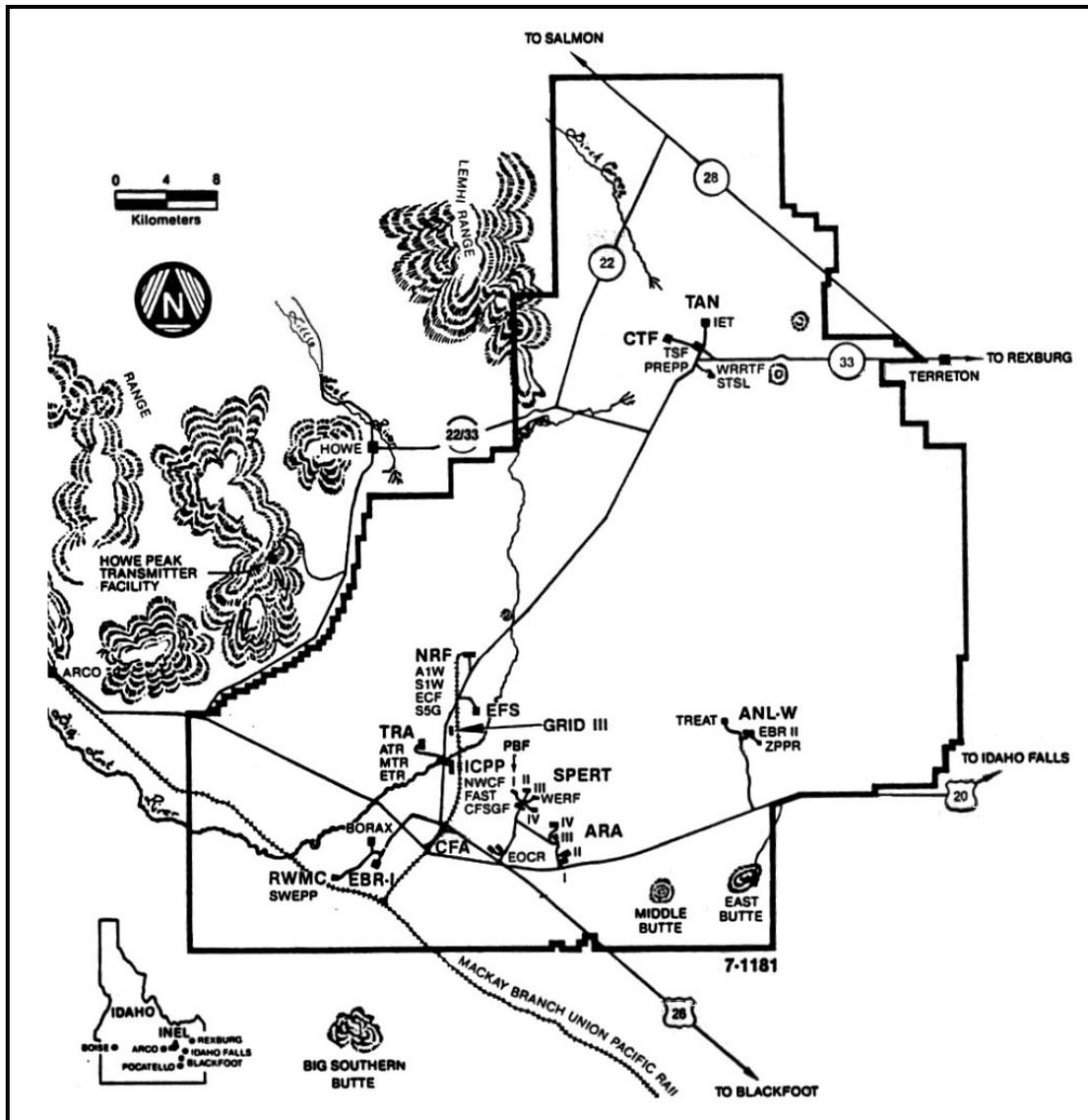
Figure 2-1. Idaho National Laboratory (Hoff 1985, 1986). Facilities include ANL-W, where the Experimental Breeder Reactor No. 2 (EBR-II), Transient Reactor Experiment and Test (TREAT) Facility, and Zero Power Plutonium (later Physics) Reactor (ZPPR) are located; Army (later Auxiliary) Reactor Area (ARA); 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 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 at TAN.

In cases where airborne radioactivity might be present or internal exposure was possible, respirators were provided to prevent/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, or releases to onsite personnel and to members of the public. Each facility has had film or TLD badges in specific building areas and around perimeter fences to measure direct environmental radiation accumulation at the location as a check and balance on source terms in site locations [4].

In addition to the nuclear reactor experiments discussed below in relation to the TRA, 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.

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

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.

2.2   TEST AREA NORTH

TAN is 30 mi northeast of the CFA. General Electric Company (GE) opened TAN in 1952 for the Aircraft Nuclear Propulsion (ANP) Program, which operated during the 1950s and early 1960s. 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/Hot Cells Annex, 1955 to 2007

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 involved in the examination, handling, analysis, or disassembly of radioactive components.

The Warm Shop, immediately adjacent to the Hot Shop, is one large open room measuring 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, or 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 involving radiation levels measured up to 50 rad/hr for brief periods have occurred in the facilities, including disassembling the Stationary Low-Power Reactor (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 and activation products associated with reactor operation (Table 2-3) [6].

External exposure exists from mixed fission products and mixed activation products (MAPs). Radiation levels varied from background levels to measured levels greater than 50 rad/hr, principally gamma rays with energy greater than 250 keV [7].

2.2.1.2   TAN 607 Storage Pool, 1955 to 2007

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. 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, including 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 in 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 mixed fission products (MFPs), activation products, 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).

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

External exposure exists from the nuclides listed above.

2.2.1.3   Storage Pads TAN 790 and TAN 791, 1986 to 2005

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). 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 alarm if either deviates from accepted limits. 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 2007

A radioactive liquid waste system (TAN 666), which was built in 1959, collected and processed intermediate-level liquid wastes generated in 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 per year (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 activation products (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/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 disturbing contaminated soil areas containing MFPs and/or activation products [14].

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

2.2.1.6   TAN 607 Radiography Facility

The TAN 607 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, or equipment. Some of the isotopic sources were in the 100-Ci range when first purchased [16]. Radiography was performed by trained and certified technicians according to 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 associated with radiographic sources [18].

2.2.2   Initial Engine Test Facility, 1955 to 1966

The IET (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 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 involving three separate reactor assemblies – HTRE 1, 2, and 3 (DOE 1991a). Jet engines were fitted to a HTRE reactor at TSF and transported by the four-rail system to the IET facility.

During the early 1960s, the AEC initiated another nuclear safety program at IET, called Systems for Nuclear Auxiliary Power (SNAP) Transients (SNAPTRAN). The program evaluated the hazards associated with 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, with the last two ending in destruction of the SNAP 10A/2 reactors.

The test series involved a number of power reactor operations that resulted in the release of radionuclides to the environment. The dates and times of reactor operations and consequent releases varied throughout the duration of the project. Therefore, the meteorological conditions that existed depended on the time of the tests. All operations were under strict meteorological control, and 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).

Internal exposure was possible from airborne MFPs and activation products. 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, p154).

External exposure occurred from the activation and MFP inventory after tests were completed and 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. The reactor was run on the test stand for 150.8 hr at full power, 20 MW thermal, exceeding

the design requirement of 100 hr. During the first 6 hr of full-power operation, fuel element damage occurred in three cartridges 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).

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 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 hr 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; Stacy 2000).

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 anticipating final design in an airframe. HTRE-3 operated for 126 hr, running two modified J47 jet engines at power as high as 32 MW thermal. This reactor was water-cooled and 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).

On November 18, 1958, a nuclear excursion occurred during IET No. 13. The excursion was the result of the dynamic and shim rods being 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 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 hr after the incident, measured 0.8 to 2.0 mrem/hr at contact roughly 1-1/2 mi from IET (DOE 1991a).

After each HTRE test, the cooling water was drained and replaced with mercury to provide shielding to personnel working on the reactors (Stacy 2000). 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.

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 activation products to the outdoor environment. In some cases the cooling air flow 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, minimizing their exposure potential.

External exposure occurred from the MFP and activation product inventory during work associated 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 hazards associated with using nuclear reactors for aerospace power systems. The portion of the program concerned with determining the kinetic behavior of the SNAP 10A/2 reactors and the consequences of certain nuclear accidents involving these reactors was called SNAPTRAN. The tests were conducted at the IET facility at TAN. The SNAPTRAN program extended the SPERT reactor safety testing program (see Section 2.9) to aerospace applications. Three test series involving 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 approaching but not resulting in damage to the zirconium-hydride-uranium fuel (Stacy 2000). The SNAP 10A/2 reactors were 9 in. in diameter by 12 in. long and were composed of a sodium/potassium (NaK)-cooled core containing 37 rods of fully enriched uranium in a zirconium-hydride matrix (Cordes et al. 1967).

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 reactor (SNAP 10A/2) designed to supply auxiliary power in space. The test, conducted at the IET facility on April 1, 1964, 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 building up a high inventory of radioactive fission products. The test involved reactor operation at a power level of 30,000 MW for 1.5 ms. 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).

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

This test version of the small space reactor, SNAP 10A/2, 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, releasing 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. The reactor core was 93% enriched fuel, containing 4.75 kg of ^{235}U. The reactor operated at a power level of 36,000 MW for 1.5 ms. 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 3 hr after the tests. Wind had to be southwest (180° to 240°) between 10 and 30 mph (Cordes et al. 1967).

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 totally destroyed, releasing 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 constructed 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. 262).

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 undated). No radiological exposure resulted from the blowdown tests [22].

Due to the electrically heated experiment cores in the facilities discussed in the following sections, internal and external exposure from neutron reactions in the reactors did not exist [23]. Exposures will be noted as applicable at the end of the pertinent test series.

2.2.3.1   Shield Test Pool Facility

The STPF, initially used for a reactor experiment known as Susie, was used for bulk shielding experiments performed 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; 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 MW thermal. After the ANP Program ended in 1961, Susie was used by other programs at INL (Walsh 1961). The facility 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, as well as instrument calibrations. 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 exposures 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 (IC) 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 IC 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.

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 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 intended for testing in HTRE-2 were first evaluated for reactivity characteristics in the CET. The water-moderated CET utilized a beryllium reflector (Becar et al. 1961).

CET was one of three low-power reactors supporting the ANP Program, along with the STPF Reactor (Susie) and the Hot Critical Experiment (HOTCE) (Hoefer 1957). The ANP Program ended in 1961.

2.2.3.2.2 Hot Critical Experiment, 1958 to March 28, 1961

Located in the LPTF CE cell, HOTCE was an elevated-temperature critical experiment 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 separated. The normal operating power was 1 W for a period of 1 to 3 hr. 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 nuclear 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 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 W thermal not to exceed 1,000 W 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. 263).

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 started by the National Aeronautics and Space Administration (NASA) at LPTF to investigate the propulsion of space rockets by nuclear power, offering 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 being 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. 261).

2.2.3.2.6 Spherical Cavity Reactor Critical Facility, 1972 to November 1973

The Spherical Cavity Reactor Critical Facility (SCRCE) was the final experiment in reactor physics work for the NASA-sponsored 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. The SCRCE assembly consisted of two aluminum tanks, one inside the other, with D_2O in the space between the two tanks. The D_2O would act 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 operated 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. 261).

Internal exposure was not known to occur [28].

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

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 pressurized-water power plant 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 loss-of-coolant accident (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, following completion of the LP-FP-2 experiment, 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. 264).

The LOFT facility was built 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 CAM, and respiratory protection was worn as required. On July 9, 1985, following completion of the LP-FP-2 test, leakage was discovered from the fission product monitoring system and the primary coolant system, allowing fission products to enter the reactor building. Over the following 2-mo 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; Author unknown 1985).

External exposure occurred to personnel working inside the containment vessel or on the primary system or sample systems associated with the reactor. During initial entry after a test, the fields in containment were ≥ 100 mrad/hr beta-gamma. The short-lived fission products would die off rapidly, reducing the general fields to ≤ 10 mrad/hr beta gamma [30].

2.2.5   Specific Manufacturing Capability, 1985 to 2007

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, located in TAN 607A and the TAN Hazardous Waste Storage Area, located 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).

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 2007

INTEC, formerly the Idaho Chemical Processing Plant (ICPP or CPP), commonly known as the "Chem Plant", is 53 mi west of Idaho Falls and occupies 200 acres in the middle of the INL reservation. The Plant 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 (DOD) projects. INTEC received the first fuel shipment in November 1951. The first hot run started in February 1953. Reprocessing continued until the fuel reprocessing project was cancelled in 1992.

The primary INTEC mission involved reprocessing spent nuclear fuel with highly enriched uranium, which entailed extracting reusable uranium from spent fuels. Each cladding (e.g., aluminum, zirconium, stainless-steel, and graphite) or fuel type called for a different process. The numerous fuel recycling processes required support facilities for fuel dissolution and recovery of fissionable materials (e.g., 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 generated during 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 a 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 that resulted 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 and/or leaking systems containing mixed fission products, TRU materials, activation products, 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 601/602, February 1953 to 1992

The INTEC Fuel Processing Facility (FPF), CPP 601/602, was used for the chemical separation of highly enriched uranium (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, producing uranyl nitrate and 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 and extends to 57 ft below grade and 38 ft above grade and features at least seven corridors for different functions (Cederberg et al. 1974). Table 2-4 lists the cells and processes associated with the fuel reprocessing. With the exception of the Health Physics (HP) 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 results in many activities being conducted with significant dose rates and/or contamination levels [35].

Table 2-4. ICPP 601/602 process cell information (Cederberg et al. 1974)

Cell	Process description	Cell function
A	EBR feed preparation	Fuel dissolution
C	MTR feed preparation	Fuel dissolution
D	MTR feed preparation	Fuel dissolution

E	STR-(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	First-cycle extraction	
Q	Second-cycle extraction	
R	Product transfer cell	
S	Third-cycle extraction	
T	Solvent pumps	Hexone storage
U	First-cycle aqueous raffinate treatment	
V	HP field office	
W	First-cycle solvent raffinate	
X	Sample dilution and decontamination	
Y	Second- and third-cycle raffinate treatment	
Z	Product room	

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 exceeding 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.

In 1956, a process for the recovery of radioactive ^{140}La (RaLa) was assigned to ICPP. 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-d half-life. The RaLa process recovered ^{140}Ba for its ^{140}La daughter product and shipped it to Los Alamos National Laboratory for use in weapon research projects [36].

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 6.6 kg of neptunium was shipped to the Savannah River Site (SRS) 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 of CPP-627. The process did not separate the neptunium, so it remained a minor constituent of the product and waste [37].

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. FPR was about 40% complete when construction stopped in 1992. FPR was 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 containing 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

containing about 600 to 900 L of water. The resulting criticality lasted about 20 min. 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, triggering 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 + gamma, mostly beta) (Ginkel et al. 1960).

- <u>Criticality Accident of January 25, 1961</u>. 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. 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 1961).

- <u>ICPP Criticality Accident of October 17, 1978</u>. 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, and 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 associated with the criticality event (INEL 1977, p 99).

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

<u>External exposure</u> occurred from work performed in Radiation Areas and High Radiation Areas containing the nuclides referenced in Table 2-3. Brief exposures to measured levels of 50 rad/hr existed in the cells during piping/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 (PIF; 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 PIF 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 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, making 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 mo 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. High-level waste in storage can be routed to permanent storage tanks. Intermediate-level waste can be routed either to 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 work stations there [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 provided remote handling of samples brought to the boxes. The equipment is operated by hand-operated manipulators extending 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 collected in stations in the CPP-601 sample corridor were transferred to the RAL for analysis. Frequent samples during fuel reprocessing, taken in 10-mL sample bottles, 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 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, contaminating four employees and a portion of the lab. Internal exposures ranged from 0.24 to 9.1 mrem/yr for 50 years. The unirradiated material came from a cleanup campaign at Argonne National Laboratory–East (ANL-E) and is suspected to have contained zirconium, which would react explosively with the nitric acid being used to dissolve the uranium (Decker 1991).

2.3.3   INTEC Fuel Storage Facility (CPP-603), 1950 to 2007

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 attacking the concrete pool. Anticontamination apparel was provided, but generally respiratory protection was 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 stored there came from the Fort St. Vrain commercial reactor in Colorado. Shipments from Fort St. Vrain ended in 1991.

In the 1950s, the ICPP received a request to process spent fuel elements from SRS reactors. The 14-ft elements were clad in aluminum and had to be cut to 18-in. lengths to fit in the dissolver vessel. Irradiation of the fuel changed metallurgical characteristics, so rather than cutting like regular aluminum they crumbled, necessitating 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, once in underwater storage at Building 603, has been transferred to the newer underwater storage pools at the Fluorinel Dissolution Process and Fuel Storage Facility (FAST) or to dry storage. The facility is being evaluated for D&D. As part of this, the CPP-603 storage pools were cleaned of sludge, dewatered, and filled with grout in 2006 (Rockhold 2007).

Internal exposure potential was greatest from fuel cutting and shearing, which created airborne radioactivity from the cut SRS 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 2007

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 generated during 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, reducing the volume and converting 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. However, 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.

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 TLDs 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 generated 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 ground-water 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 created during flushing operations, valve repairs, or other maintenance activities in contamination areas from MFPs and activation products [44].

External exposure occurred during work in the contamination areas and on valve changes and piping maintenance where exposure rates measured 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 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 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. 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 raising the bed temperature to 500°C, reducing wall temperatures and ruthenium concentrations in the off-gas.

In 1982, the New Waste Calcining Facility (NWCF) replaced WCF-1. It converted liquid high-level radioactive waste from the Tank Farm into a granular solid similar in consistency to sand. The liquid waste was drawn from underground storage tanks and sprayed into a calciner vessel 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 encased in thick concrete vaults called bin sets. The calciner was shut down in May 2000 while DOE evaluated whether to upgrade it to meet new emissions standards or 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].

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 NWCF was the location of a decontamination facility used 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 would create 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 2007

FAST has two parts: a spent fuel storage area and the Fluorinel Dissolution Facility (FDF). The storage area consists of six stainless-steel-lined pools for storing spent nuclear fuel. The FDF includes a hot cell with 6-ft-thick concrete walls where spent fuel was dissolved in an acid solution. The Fluorinel process was used on Zircaloy-clad naval fuels and used three dissolver-complexer trains operating batchwise. Soluble neutron poisons and limiting the 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).

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 2007

The INL Comprehensive Environmental Response, Compensation, and Liability Act (CERCLA) Disposal Facility (ICDF) Complex is a new engineered facility south of INTEC and adjacent to the existing percolation ponds. This facility was planned to begin operation in July 2003 (INEEL 2003). It

is designed and authorized to accept wastes from INL CERCLA actions. The ICDF 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 Resource Conservation and Recovery Act (RCRA) Corrective Action Management Unit and is the disposal site for leachate and other aqueous wastes generated as a result of operating the ICDF 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 radioactive materials consisting of MFPs and activation products 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 2007

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 is the site of the now-decommissioned EBR-I, Boiling Water Reactor (BWR) Experiment, Argonne Fast Source Reactor (AFSR), and 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, DOE reorganized; ANL-W became part of the Idaho National Laboratory (INL), and that portion of the site became the Materials and Fuels Complex (MFC).

2.4.1   Experimental Breeder Reactor No. I, April 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.

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, $\tau_{1/2}$ = 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 (Haroldson 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, Late 1953 to July 22, 1954

The date information in this discussion is from Stacy (2000, p. 261). 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. 261). 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 [61].

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

The date information in this discussion is from Stacy (2000, p. 261). 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. BORAX-III became the first nuclear power plant in the world to generate electricity for an entire city. It was operational from June 9, 1955, until sometime before December 3, 1956.

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 principle source in the coolant (Dietrich, Lichtenberger, and Zinn 1956).

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

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 atmosphere 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. 261). 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. 268). 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 1970

AFSR 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 experiment, was built around a cylindrical core of solid HEU with a blanket of solid DU (Brunson 1959). 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, 1958, to April 1994

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

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

<u>External exposure</u> occurred from routine operations [73].

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

The date information in this discussion is from Stacy (2000, p. 262). 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 is 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 2007

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, 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.

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 irradiated in EBR-II. 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 conducted in HFEF provide data that are essential for determining the performance and condition of fuels and materials 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 activation products [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. 268). 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 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 are present between the two halves when the reactor was off and open (Simons, Young, and Thalgott 1972). The control and safety rod drives are 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, is 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 is used primarily to dry and coat depleted uranium 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 2006

The Neutron Radiography (NRAD) Facility is a 250-kW, steady-state Training, Research, Isotope General Atomics (TRIGA) reactor in the basement beneath the HFEF/North main cell. The reactor core consists of fuel elements surrounded by graphite assemblies. The core is submerged in a water-filled tank. NRAD began operation on October 12, 1977, 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 a 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 activation products [78].

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

2.4.14   Fuel Assembly and Storage Building, 1970 to 2007

The Fuel Assembly and Storage Building (FASB) 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, etc. The West room contains a vault for the storage of nuclear material. It also contains equipment for performing materials testing and for preparing metallurgical samples, and inert atmosphere gloveboxes and hoods. The facility no longer does fuel assembly, but other radiological work is ongoing.

Internal exposure might occur from airborne radioactivity associated with 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. The facility includes six analytical hot cells and general and specialized chemistry laboratories. Personnel were subjected to radiation levels above building background 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 generated at ANL-W facilities into solidified residue packaged in shielded containers.

The Industrial Waste Pond (IWP) 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 going to the IWP 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 (STPs) 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 associated with the types of samples worked in these facilities [82].

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

2.5   RADIOACTIVE WASTE MANAGEMENT COMPLEX, MAY 1952 TO 2007

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.

Early packaging configurations were thought to be not suitable 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 waste. 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. There were occasional brief exposures greater than 100 mrem/hr (photon energy ≥ 250 keV).

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

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 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).

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 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 Transuranic Storage Area (TSA) is in the southern section of the RWMC, 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.

2.5.4   Stored Waste Examination Pilot Plant

The Stored Waste Examination Pilot Plant (SWEPP) certifies waste to ensure that it meets repository acceptance criteria. Examinations used a shielded 450-kVp X-ray facility and a neutron assay system that developed 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.

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 (BNFL), Inc.

2.5.6   Advanced Mixed Waste Treatment Project

BNFL operates the Advanced Mixed Waste Treatment Project under contract with DOE. The facility, which is under construction, 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.

The exposures below represent all subsections of Section 2.5.

Internal exposure might have resulted when workers were close to breached and or leaking waste containers containing MFPs, TRU materials, activation products, 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 SWEPP X-ray (30 to 450 keV) and neutron (2 to 20 MeV) waste package examination and certification processes. Approximately 150 6M drums containing as much as 500 g of $^{232/233}$U per drum have been stored under earthen covers on pads in the 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 involved, 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].

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. Activation products from reactor facilities were also a concern for external exposure.

2.6   CENTRAL FACILITIES AREA, AUGUST 2, 1943, 2007

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 (R&D) 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 from fighting ships. Buildings constructed by the Navy became the staging area for INL development that began in earnest in 1950. The area continued to expand as a central service area for the NRTS. Eventually it acquired the name "Central," or officially CFA.

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 exposures discussed below are for all subsections of Section 2.6.

Internal exposure. CFA internal exposure potential was primarily from MFPs, activation products, and TRU materials associated with articles 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 working in radiological areas must wear dosimetry devices [88].

2.6.1   Hot Laundry

The laundry, in the east portion of the CFA, washed coveralls and other protective clothing items used for 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 for 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 used for radiological instrumentation standardization. The HPIL used ^{252}Cf neutron sources, and alpha, beta, and gamma sources for HP instrument calibrations. All the sources were sealed [89]. A new facility, CF-1618, completed in late 2002, 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 [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 used by DOE-ID 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 RESL facility, operated by DOE-ID 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, TLDs, etc.

2.6.5   CF-674 Building

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

2.7   TEST REACTOR AREA, MARCH 31, 1952, TO 2007

The TRA is approximately 5 mi north of the CFA. Eight reactors have been built and operated in the TRA. Three of the reactors – Materials Test Reactor, Engineering Test Reactor (ETR), and Advanced Test Reactor (ATR) – were high-flux reactors designed 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 designed to perform reactivity measurements. At present, only the ATR and ATRC are operational.

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 entering the TRA must wear a dosimetry badge, and those who work in or near radiological control areas must wear PICs. Personnel working 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. 264). The MTR (TRA 603) was the original reactor at the TRA and the second reactor to be operated at INL. Fueled with enriched uranium fuel, water-cooled and water-moderated, this reactor was a key part of the AEC postwar reactor development program. It operated at a power level of 30 MW until September 1955, when thermal output was increased to 40 MW. It supplied a high neutron flux in support of a reactor development program subjecting potential reactor fuels and structural materials to irradiation.

The MTR gave the researcher 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 discharge 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-hr run to irradiate 1,000 biological samples for iodine analysis [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 activation products from maintenance activities resulted in the potential for internal exposure. Some experiments in loops resulted in releases, particularly of activation products [94].

External exposure occurred during the operations associated with sampling 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. 262). 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.

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 exceeding 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 7, 1967, to 2007

The date information in this discussion is from Stacy (2000, p. 260). 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, making 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-radiation 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 surrounding the loops enable additional tests. In addition, the ATR produces radioisotopes for use in medicine, industry, and other research.

Internal exposure might have occurred during reactor shutdown from mixed fission and activation products 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 exposures are 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 depending on the nuclide and quantity. Typical nuclides generated would 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 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).

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

External exposures were 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.

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

The date information in this discussion is from Stacy (2000, p. 260). 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.

Internal exposure. 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 (CFRMF). 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 CFRMF contributed to the development of fast neutron reactors (Stacy 2000).

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

External exposure. Minimal exposure resulted 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, saving time and money (Stacy 2000).

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

External exposure. Irradiated fuel or samples were transferred 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 activation products during loading or unloading fuel or test samples [107].

2.7.9 Advanced Test Reactor Critical Facility, May 19, 1964, to 2007

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).

Internal exposure was possible from fission products released from damaged fuel or experiment samples containing 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, pg. 9].

2.7.10 TRA Hot Cell Facility, 1954 to 2007

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. 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 performed 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 mixed fission and activation products 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 (Stacy 2000).

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.

Internal exposure was possible from a spent fuel element 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 2007

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 brought in for counting. The isotopes encountered would include fission products, activation products, TRU materials, and more [113].

External exposures are very low due to the small samples 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 R&D 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, 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 exposures occur 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. The seepage ponds 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 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 HP 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 ICPP 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 activation products [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 units. The ARA reactors were built and maintained by contractors with a mixed cadre of military personnel training on the operation of the facilities. ARA is 10 mi east of CFA; it began with the ARA-I site one-half-mile north of Highway 20. ARA-II, -III, and -IV are at half-mile intervals along an access road, Fillmore Avenue, north from Highway 20. After the Army phased out its program around 1965, ARA facilities were used for experiments and tests involving multiple radionuclides, particularly at the ARA-I hot cell and laboratory facilities (Stacy 2000). D&D of the ARA ended in 2002.

2.8.1 ARA-I

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/SL-1, February 23, 1960, to April 6, 1961

ARA-II, a half-mile north of ARA-I, was the site of a low-power BWR designed and built by ANL based on its BORAX experience. The reactor, designed to generate only 1,000 kW, was originally named the Argonne Low Power Reactor. 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 between 1 and 6 wk, and then would be shut down for scheduled maintenance and test changes.

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, creating a steam explosion that killed three persons and destroyed the reactor. 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 exceeding 500 rad/hr to personnel in radiation fields. Extensive cleanup efforts followed, including 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 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

ARA-III, another half-mile 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 kW of heat, but no electricity. The Army wanted to develop a mobile nuclear power plant, and the GCRE was the first phase of that program, proving 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. 263).

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

ARA-IV, another half-mile 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 weighing 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-hr 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 around 1965.

The following exposures relate to all ARA facilities (Stacy 2000, p. 265).

Internal exposure potential was typical of that associated with 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 consisting of MFPs and activation products would cause most internal exposures [121].

External exposure was significant from the SL-1 excursion and recovery operations. Nine technical overexposures resulted, ranging from 15 to 27.3 rem (whole-body dose) (Horan and Braun 1993). Hot cell work and D&D efforts in the reactor areas contributed to exposure pathways. Hot cell entries after working with irradiated experiments were a source of high exposure. Expected exposure potential was from photons greater than 250 keV from the MFPs and activation products associated

with 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].

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

The Waste Reduction Operations Complex/Power Burst Facility (PBF)/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 waste, and research to reduce hazardous and mixed waste.   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. Internal and external exposure will be noted at the end of Section 2.9 for the following subsections.

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

SPERT-I was an open-tank, light-water-moderated, reflected reactor, originally using 92% enriched uranium fuel. The reactor tank, about 4 ft in diameter and 14 ft high, was the first in a series of four safety-testing reactors 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. 266).

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, with partial melting in approximately 35% of the core. 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 a half-mile from the reactor. SPERT-I tests demonstrated the damage-resistant capabilities of low-enrichment (4% ^{235}U) uranium-oxide fuel pins similar to those used in water-cooled reactors powering large central stations.

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 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 (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. 265).

At present, the area is used as a lead storage facility; it has been primarily used for storage since 1964. 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 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 used in commercial power production. Water could flow through the vessel at a rate as high as 20,000 gal/min, handle temperatures as high as 650°F, and pressures as high as 2,500 psi (Stacy 2000, p. 266).

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 varying conditions including forced coolant flow, variable 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 (CDC), 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 "driver" fuel in the rest of the core. Work on fuel destructive mechanisms continued until the PBF replaced the CDC.

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. But 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. 266).

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 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 severe-fuel-rod-burst tests 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 codes 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. 266).

The PBF was a high-performance, water-cooled, uranium-oxide-fueled reactor 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 the ICPP. 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 [123].

2.9.7   Waste Experimental Reduction Facility, 1982 to 2001

The Waste Experimental Reduction Facility (WERF) 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 (radioactive and hazardous) waste. Wastes from INL and other DOE facilities were treated under provisions in the INEEL Site Treatment Plan (Stacy 2000). INL began the closure process for the WERF, which should take 3 years, in 2001.

2.9.8   Mixed Waste Storage Facility

The Mixed Waste Storage Facility, in the 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 stored in the facility.

Internal exposures were 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, making internal exposure possible, with ^{137}Cs being the primary radionuclide [124].

External exposure resulted from experiment changes related to reactor 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 activation products [125].

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

The Organic-Moderated Reactor Experiment (OMRE), a few miles east of the CFA, was built 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, significantly reducing the risk of leaks. However, it lacked test loops needed to investigate various organic coolants and experimental fuel elements (Stacy 2000, p. 266).

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 recycled for other (non-nuclear) uses.

Internal Exposure. On November 16, 1960, an experiment was conducted to determine the feasibility of open-air burning of contaminated solvents 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 exposures were possible from MFPs and activation products 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, near Lincoln Boulevard south and east of NRF and east of the TRA and INTEC, was the site of several tests with atmospheric releases. It originated in the 1957 timeframe, and was based on concerns about what would happen from nuclear aircraft crashes, etc. This highly instrumented grid would measure atmospheric conditions 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. Exposure information is summarized at the end of Section 2.11 for the following subsections.

2.11.1 Fuel Element Burn Tests

Fuel Element Burn Tests A and B were conducted on Grid III to support the GE ANP Program to evaluate the consequences of a nuclear aircraft crash involving 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 following 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) containing 5,000 Ci of fission products. A pool of jet fuel was ignited under the fuel element, which reached a temperature of about 2,250°F. After the fire, the fuel element was intact with a small puncture in the cladding [128].

FEBT-B used an induction furnace that heated a fuel element containing 10,000 Ci of fission products to 5,000°F. Most of the fuel element melted and dispersed within 90 s 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).

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 involving nuclear-powered aircraft. Nine tests evaluated release percentages, airborne radioactivity, and diffusion and deposition characteristics of fission products released from melted aircraft reactor fuel

elements (Convair 1959). Five tests were with fuel decayed for 922 to 985 d and four were with fuel decayed for 42 to 65 d. Operating temperatures were between 1,000 and 2,300°C. To simulate a potential accident, the tests used an induction-type furnace to heat the elements rapidly to the melting point in approximately 2 min, and maintained this temperature for approximately 10 min 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 ranging from 32 to 600 Ci; tests in 1970 consisted of ^{24}Na releases ranging from 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 associated with the tests (Till et al. 2002).

Internal exposures were 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 [129].

External exposure would have resulted from working in the proximity of, handling, or transporting 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 [130].

2.12 EXPERIMENTAL FIELD STATION/EXPERIMENTAL DAIRY FARM

The EFS was a 27-acre plot about 7 mi northeast of the ICPP 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 objectives of the Controlled Environmental Radioiodine Test (CERT) releases were 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 ranging in amount from 0.05 to 8 Ci. Most of the releases occurred at the Experimental Dairy Farm. Others

occurred at ICPP, ARA, NRF, and CFA. 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. Hawley (1964) reports that the CERT No. 1 release occurred near ground level over a 30-min period. Additional information regarding this test series is provided by Hawley (1964); Bunch (1966, 1968), and Zimbrick and Voillequé (1969). Early in the test program, the AEC granted permission to six DOE-ID volunteers 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, although 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 2007

The INL Research Center (IRC), a 35-acre site on North Boulevard in Idaho Falls, was built between 1982 and 1984. The facility was dedicated in 1984 to further the INL research and engineering mission. The IRC has 58 laboratories in IF-603 for geophysics, chemistry, microbiology, and other sciences; 18 of these are general-purpose modules for electronics design, optics, lasers or materials testing, and nondestructive examination research and development. The Center conducts laboratory work with tracer-level radionuclides. One of the early missions was work for the Bureau of Mines with ores, some containing natural uranium. The INL Engineering Demonstration Facility (IF-657) houses several prototypical-scale R&D 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 material 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 DOD 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 spent MTR fuel elements to evaluate the accuracy of the Radiological Safety Analysis Computer-generated code 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 NaK coolant radiologically contaminated 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 ~ 40 R/hr. The radiological fission product inventory in the NaK reported 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.

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, 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.

[1]   Frank Hinckley. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-August 2003.
        This statement is based on information gleaned from a review of the references ACC 1952 and Cipperly 1958.

[2]   Frank Hinckley. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-August 2003.
        This statement is based on information gleaned from a review of ACC 1952.

[3]   Frank Hinckley. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-August 2003.
        This statement is based on information gleaned from a review of ACC 1952.

[4]   Frank Hinckley. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-August 2003.
        This statement is based on a review of the reference Hoff et. al. 1986.

[5]   Don Marshall. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-August 2003.
        This statement is based on a review of the reference Stacy 2000.

[6]   Don Marshall. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-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 mixed fission products due to the fissioning of the uranium and mixed activation products due to activation from neutron bombardment. It is reasonable to expect that the potential for internal exposure to mixed fission products and mixed activation products exists in a facility that handles reactor fuel and components.

[7]   Don Marshall. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-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 the ORAUT-TKBS-0007-6 INL – Occupational External Dosimetry TBD.

[8]   Don Marshall. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-August 2003.
        This information is based on understanding the safety program and sources of exposure at TAN.

[9]   Don Marshall.   Principal Health Physicist. Intrepid Technology and Resources, Inc. June-August 2003.
        This information is based on understanding the safety program at TAN and work experience at the facility.

[10]   Don Marshall. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-
        August 2003.
        Should one of the casks leak, internal exposures could occur as the casks contain spent
        nuclear fuel and are pressurized.

[11]   Don Marshall. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-
        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. Intrepid Technology and Resources, Inc. June-
        August 2003.
        The liquid waste stream contained mixed fission and mixed activation products. 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. Intrepid Technology and Resources, Inc. June-
        August 2003.
        This information is based on a review of the reference Bonney et. al. 1995.

[14]   Don Marshall. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-
        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
        the ORAUT-TKBS-0007-5 INL – Occupational Internal Dose TBD.

[15]   Don Marshall. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-
        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 the ORAUT-TKBS-0007-6 INL – Occupational External
        Dosimetry TBD.

[16]   Don Marshall. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-
        August 2003.
        This information is based on work experience at TAN.

[17]   Don Marshall. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-
        August 2003.
        This information is based on work experience at TAN.

[18]   Don Marshall. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-
        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 the ORAUT-TKBS-0007-6 INL – Occupational External
        Dosimetry TBD.

[19] Don Marshall. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-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 the ORAUT-TKBS-0007-6 INL – Occupational External Dosimetry TBD.

[20] Don Marshall. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-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 the ORAUT-TKBS-0007-6 INL – Occupational External Dosimetry TBD.

[21] Don Marshall. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-August 2003.
This information is based on understanding the sources of exposure at TAN and a review of the reference Cordes et al. 1965.

[22] Don Marshall. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-August 2003.
The tests were performed with nonradioactive materials.

[23] Don Marshall. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-August 2003.
This information is based on understanding the sources of exposure at WRRTF and work experience at the facility.

[24] Don Marshall. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-August 2003.
This information is based on work experience at TAN.

[25] Don Marshall. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-August 2003.
This information is based on understanding the sources of exposure at TAN.

[26] Don Marshall. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-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 the ORAUT-TKBS-0007-6 INL – Occupational External Dosimetry TBD.

[27] Don Marshall. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-August 2003.
This information is based on a review of the reference Kunze and Chase 1970.

[28] Don Marshall. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-August 2003.
This information is based on understanding the sources of exposure at TAN.

[29] Don Marshall. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-
      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 the ORAUT-TKBS-0007-6 INL – Occupational External Dosimetry TBD.

[30] Don Marshall. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-
      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 the ORAUT-TKBS-0007-6 INL – Occupational External
      Dosimetry TBD.

[31] Don Marshall. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-
      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 the ORAUT-TKBS-0007-5 INL – Occupational Internal Dose
      TBD.

[32] Don Marshall. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-
      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 the ORAUT-TKBS-0007-6 INL – Occupational External Dosimetry TBD.

[33] Boyd Leavitt. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-
      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 the ORAUT-TKBS-0007-5 INL – Occupational Internal Dose
      TBD.

[34] Boyd Leavitt. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-
      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 the ORAUT-TKBS-0007-6 INL – Occupational External
      Dosimetry TBD.

[35] Boyd Leavitt. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-
      August 2003.
      This information is based on understanding the sources of exposure at ICPP and work
      experience at the facility and a review of the reference Cederberg et. al. 1974.

[36] Boyd Leavitt. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-
      August 2003.
      This information is based on a review of the reference Stacy 2000.

[37] Boyd Leavitt. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-
      August 2003.
      This information is based on understanding the sources of exposure at ICPP and work
      experience at the facility.

[38]   Boyd Leavitt. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-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 the ORAUT-TKBS-0007-5 INL – Occupational Internal Dose TBD.

[39]   Boyd Leavitt. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-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 the ORAUT-TKBS-0007-6 INL – Occupational External Dosimetry TBD.

[40]   Boyd Leavitt. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-August 2003.
       This information is based on understanding the sources of exposure at ICPP and work experience at the facility.

[41]   Boyd Leavitt. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-August 2003.
       This information is based on a review of the reference Stroschein and Maeser 1967.

[42]   Boyd Leavitt. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-August 2003.
       This information is based on understanding the sources of exposure at ICPP and 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 the ORAUT-TKBS-0007-6 INL – Occupational External Dosimetry TBD.

[43]   Boyd Leavitt. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-August 2003.
       This information is based on work experience at the Tank Farm.

[44]   Boyd Leavitt. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-August 2003.
       This information is based on understanding the sources of exposure at the Tank Farm and work experience at the facility. This internal exposure scenario is routinely included in safety analyses of this type of facility, as described in section 5.7.2.2 of the ORAUT-TKBS-0007-5 INL – Occupational Internal Dose TBD.

[45]   Boyd Leavitt. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-August 2003.
       This information is based on understanding the sources of exposure at the Tank Farm and 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 the ORAUT-TKBS-0007-6 INL – Occupational External Dosimetry TBD.

[46]   Norm Rohrig. Consulting Health Physicist. Intrepid Technology and Resources, Inc. April 25, 2007.
       ERDA 1977 p 106 shows gross beta activity of 4840 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. Intrepid Technology and Resources, Inc. June-
        August 2003.
        This information is based on work experience at the WCF.

[48]   Boyd Leavitt. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-
        August 2003.
        This information is based on understanding the sources of exposure at the WCF and work
        experience at the facility. This internal exposure scenario is routinely included in safety
        analyses of this type of facility, as described in section 5.7.2.3 of the ORAUT-TKBS-0007-5
        INL – Occupational Internal Dose TBD.

[49]   Boyd Leavitt. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-
        August 2003.
        This information is based on understanding the sources of exposure at the WCF and 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 the ORAUT-TKBS-0007-6 INL
        – Occupational External Dosimetry TBD.

[50]   Boyd Leavitt. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-
        August 2003.
        This information is based on understanding the sources of exposure at the FDF and work
        experience at the facility.

[51]   Boyd Leavitt. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-
        August 2003.
        This information is based on understanding the sources of exposure at the FDF and work
        experience at the facility.

[52]   Boyd Leavitt. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-
        August 2003.
        This information is based on understanding the sources of exposure at the ICDF and 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 the ORAUT-TKBS-0007-6 INL
        – Occupational External Dosimetry TBD.

[53]   Boyd Leavitt. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-
        August 2003.
        This information is based on understanding the sources of exposure at the ICDF and work
        experience at the facility.

[54]   Boyd Leavitt. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-
        August 2003.
        This information is based on understanding the sources of exposure at the ISFSI and work
        experience at the facility.

[55]   Frank Hinckley. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-
        August 2003.
        This information is based on understanding the sources of exposure at the ISFSI and 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 the ORAUT-TKBS-0007-6 INL
        – Occupational External Dosimetry TBD.

[56]   Graham, Earl. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-August 2003.
       Information provided is based on a review of the reference Haroldson et. al. 1963.

[57]   Graham, Earl. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-August 2003.
       Information provided is based on a review of the reference Haroldson et. al. 1963.

[58]   Graham, Earl. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-August 2003.
       Information provided is based on a review of the references Dietrich et. al. 1956 and Griffiths et. al. 1956.

[59]   Graham, Earl. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-August 2003.
       Information provided is based on a review of the references Dietrich et. al. 1956 and Griffiths et. al. 1956.

[60]   Graham, Earl. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-August 2003.
       Information provided is based on a review of the references Dietrich et. al. 1956 and Griffiths et. al. 1956.

[61]   Graham, Earl. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-August 2003.
       Information provided is based on a review of the references Dietrich et. al. 1956 and Griffiths et. al. 1956.

[62]   Graham, Earl. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-August 2003.
       Information provided is based on a review of the reference Zinn et. al. 1956.

[63]   Graham, Earl. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-August 2003.
       Information provided is based on a review of the reference Zinn et. al. 1956.

[64]   Graham, Earl. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-August 2003.
       Information provided is based on a review of the reference Robertson and Hall 1959.

[65]   Graham, Earl. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-August 2003.
       Information provided is based on a review of the reference Robertson and Hall 1959.

[66]   Graham, Earl. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-August 2003.
       Information provided is based on understanding of the sources of exposure based on experience at BORAX-V.

[67]   Graham, Earl. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-August 2003.
       Information provided is based on understanding of the sources of exposure based on experience at BORAX-V.

[68]   Graham, Earl. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-August 2003.
       Information provided is based on a review of the reference Brittan et. al. 1961.

[69]   Graham, Earl. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-August 2003.
       Information provided is based on a review of the reference Brittan et. al. 1961.

[70]   Graham, Earl. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-August 2003.
       Information provided is based on understanding of the sources of exposure based on experience at AFSR and a review of the reference Brunson 1959.

[71]   Graham, Earl. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-August 2003.
       Information provided is based on understanding of the sources of exposure based on experience at AFSR and a review of the reference Brunson 1959.

[72]   Graham, Earl. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-August 2003.
       Information provided is based on a review of the reference Freund et. al. 1960.

[73]   Graham, Earl. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-August 2003.
       Information provided is based on a review of the reference Freund et. al. 1960.

[74]   Graham, Earl. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-August 2003.
       This statement by Mr. Graham is based on his understanding of the sources of exposure based on experience at HFEF. Since the hot cells were used to examine irradiated fuel, they would be contaminated with mixed fission and activation products. This contamination is a potential internal exposure hazard during hot cell entries.

[75]   Graham, Earl. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-August 2003.
       This statement by Mr. Graham is based on his understanding of the sources of exposure based on experience at HFEF. Since the hot cells were used to examine irradiated fuel, they would be contaminated with mixed fission and activation products. This contamination is a potential external exposure hazard during hot cell entries.

[76]   Graham, Earl. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-August 2003.
       Information provided is based on understanding of the sources of exposure based on experience at ZPPR and a review of the reference Kunze and Chase 1970.

[77] Graham, Earl. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-August 2003.
Information provided is based on understanding of the sources of exposure based on experience at ZPPR and a review of the reference Kunze and Chase 1970.

[78] Graham, Earl. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-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 mixed fission and activation products. This contamination is a potential internal exposure hazard.

[79] Graham, Earl. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-August 2003.
This statement by Mr. Graham is based on his understanding of the sources of exposure based on experience at NRAD and a review of the reference Richards and McClellan 1979.

[80] Graham, Earl. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-August 2003.
Information provided is based on understanding of the sources of exposure based on experience at FASB.

[81] Graham, Earl. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-August 2003.
Information provided is based on understanding of the sources of exposure based on experience at FASB.

[82] Graham, Earl. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-August 2003.
Information provided is based on understanding of the sources of exposure based on experience at ANL-W.

[83] Graham, Earl. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-August 2003.
Information provided is based on understanding of the sources of exposure based on experience at ANL-W.

[84] Frank Hinckley. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-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 the ORAUT-TKBS-0007-5 INL – Occupational Internal Dose TBD.

[85] Frank Hinckley. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-August 2003.
This information is based on understanding the sources of exposure at the RWMC and 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 the ORAUT-TKBS-0007-6 INL – Occupational External Dosimetry TBD.

[86]   Frank Hinckley. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-
         August 2003.
         This information is based on understanding the sources of exposure at the CFA and work
         experience at the facility.

[87]   Frank Hinckley. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-
         August 2003.
         This information is based on understanding the sources of exposure at the CFA and work
         experience at the facility.

[88]   Norman Rohrig. Consulting Health Physicist. Intrepid Technology and Resources, Inc. June-
         August 2003.
         This information is based on understanding the sources of exposure at the HPIL and 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 the ORAUT-TKBS-0007-6 INL
         – Occupational External Dosimetry TBD.

[89]   Norman Rohrig. Consulting Health Physicist. Intrepid Technology and Resources, Inc. June-
         August 2003.
         This information is based on understanding the sources of exposure at the HPIL and work
         experience at the facility.

[90]   Norman Rohrig. Consulting Health Physicist. Intrepid Technology and Resources, Inc. June-
         August 2003.
         This information is based on understanding the sources of exposure at the HPIL and work
         experience at the facility.

[91]   Norman Rohrig. Consulting Health Physicist. Intrepid Technology and Resources, Inc. June-
         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. Intrepid Technology and Resources, Inc. June-
         August 2003.
         This information is based on understanding the sources of exposure at the CFA and work
         experience at the facility.

[93]   Don Marshall. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-
         August 2003.
         This information is based on work experience at the MTR area.

[94]   Don Marshall. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-
         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 the ORAUT-TKBS-0007-5 INL – Occupational Internal Dose
         TBD.

[95]   Don Marshall. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-
         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 the ORAUT-TKBS-0007-6 INL – Occupational External Dosimetry TBD.

[96]   Don Marshall. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-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 the ORAUT-TKBS-0007-5 INL – Occupational Internal Dose TBD.

[97]   Don Marshall. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-August 2003.
This information is based on understanding the sources of exposure at ETR area and 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 the ORAUT-TKBS-0007-6 INL – Occupational External Dosimetry TBD.

[98]   Don Marshall. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-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 the ORAUT-TKBS-0007-5 INL – Occupational Internal Dose TBD.

[99]   Don Marshall. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-August 2003.
This information is based on understanding the sources of exposure at Advanced Test Reactor Area and 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 the ORAUT-TKBS-0007-6 INL – Occupational External Dosimetry TBD.

[100] Don Marshall. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-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 the ORAUT-TKBS-0007-5 INL – Occupational Internal Dose TBD.

[101] Don Marshall. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-August 2003.
This information is based on understanding the sources of exposure at MTR area and 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 the ORAUT-TKBS-0007-6 INL – Occupational External Dosimetry TBD.

[102]   Don Marshall. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-
        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 the ORAUT-TKBS-0007-5 INL – Occupational Internal Dose
        TBD.

[103]   Don Marshall. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-
        August 2003.
        This information is based on understanding the sources of exposure at the ARMF and 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 the ORAUT-TKBS-0007-6 INL
        – Occupational External Dosimetry TBD.

[104]   Don Marshall. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-
        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 the ORAUT-TKBS-0007-5 INL – Occupational Internal
        Dosimetry TBD.

[105]   Don Marshall. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-
        August 2003.
        This information is based on understanding the sources of exposure at the CFRMF and 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 the ORAUT-TKBS-0007-6 INL
        – Occupational External Dosimetry TBD.

[106]   Don Marshall. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-
        August 2003.
        This information is based on understanding the sources of exposure at the ETR Critical Facility
        and work experience at the facility. This internal exposure scenario is routinely included in
        safety analyses of this type of facility, as described in section 5.7.6 of the ORAUT-TKBS-0007-
        5 INL – Occupational Internal Dose TBD.

[107]   Don Marshall. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-
        August 2003.
        This information is based on understanding the sources of exposure at the ETR Critical Facility
        and 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 the ORAUT-TKBS-0007-
        6 INL – Occupational External Dosimetry TBD.

[108]   Don Marshall. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-
        August 2003.
        This information is based on understanding the sources of exposure at the ATR Critical Facility
        and work experience at the facility. This internal exposure scenario is routinely included in
        safety analyses of this type of facility, as described in section 5.7.6 of the ORAUT-TKBS-0007-
        5 INL – Occupational Internal Dosimetry TBD.

[109]   Don Marshall. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-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 the ORAUT-TKBS-0007-5 INL – Occupational Internal Dosimetry TBD and is supported by the reference Stroschien and Maeser 1967.

[110]   Don Marshall. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-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 the ORAUT-TKBS-0007-6 INL – Occupational External Dosimetry TBD and is supported by the reference Stroschien and Maeser 1967.

[111]   Don Marshall. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-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 the ORAUT-TKBS-0007-5 INL – Occupational Internal Dosimetry TBD.

[112]   Don Marshall. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-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 the ORAUT-TKBS-0007-6 INL – Occupational External Dosimetry TBD.

[113]   Don Marshall. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-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 the ORAUT-TKBS-0007-5 INL – Occupational Internal Dose TBD.

[114]   Don Marshall. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-August 2003.
This facility is a low background facility and highly radioactive sources/samples are not allowed in.

[115]   Don Marshall. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-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 the ORAUT-TKBS-0007-5 INL – Occupational Internal Dose TBD.

[116]   Don Marshall. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-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 the ORAUT-TKBS-0007-6 INL – Occupational External Dosimetry TBD.

[117]   Don Marshall. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-August 2003.
This information is based on a review of the references Nebeker and Lakey 1970, Till et. al. 2002 and ERDA 1977a.

[118] Don Marshall. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-August 2003.
This information is based on a review of the references Nebeker and Lakey 1970, Till et. al. 2002 and ERDA 1977a.

[119] Don Marshall. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-August 2003.
This information is based on a review of the references Nebeker and Lakey 1970 and ERDA 1977a.

[120] Don Marshall. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-August 2003.
This information is based on a review of the references Nebeker and Lakey 1970 and ERDA 1977a.

[121] Frank Hinckley. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-August 2003.
This information is based on a review of the reference Horan and Gammil 1961.

[122] Frank Hinckley. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-August 2003.
This information is based on a review of the reference Horan and Gammil 1961.

[123] Frank Hinckley. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-August 2003.
This storage area is not a radiological concern since the lead stored here is of the "low background" variety.

[124] Frank Hinckley. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-August 2003.
This information is based on a review of the references INEEL 2001 and ERDA 1977.

[125] Frank Hinckley. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-August 2003.
This information is based on a review of the reference ERDA 1977.

[126] Frank Hinckley. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-August 2003.
This information is based on a review of the references ERDA 1977 and Rielly 1998.

[127] Frank Hinckley. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-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 the ORAUT-TKBS-0007-6 INL – Occupational External Dosimetry TBD.

[128] Frank Hinckley. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-August 2003.
This information is based on a review of the references SCA, SRA, and WCC 1994 and DOE 1991a.

[129]   Frank Hinckley. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-
        August 2003.
        This information is based on a review of the references SCA, SRA, and WCC 1994 and DOE
        1991a.

[130]   Frank Hinckley. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-
        August 2003.
        This information is based on a review of the references SCA, SRA, and WCC 1994 and DOE
        1991a.

[131]   Frank Hinckley. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-
        August 2003.
        This information is based on a review of the references Bunch 1968, Hawley et. al. 1964 and
        Zimbrick and Voillequé 1969.

[132]   Frank Hinckley. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-
        August 2003.
        This information is based on a review of the references Bunch 1968, Hawley et. al. 1964 and
        Zimbrick and Voillequé 1969.

[133]   Frank Hinckley. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-
        August 2003.
        This information is based on a review of the references Bunch 1968, Hawley et. al. 1964 and
        Zimbrick and Voillequé 1969.

[134]   Frank Hinckley. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-
        August 2003.
        This information is based on a review of the reference Rielly 1998.

[135]   Frank Hinckley. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-
        August 2003.
        This information is based on a review of the reference Rielly 1998.

[136]   Frank Hinckley. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-
        August 2003.
        This information is based on a review of the reference Rielly 1998.

[137]   Frank Hinckley. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-
        August 2003.
        This information is based on a review of the references Mobley and Keller 1991, Thiel 1997,
        and Mobley 1987.

[138]   Frank Hinckley. Principal Health Physicist. Intrepid Technology and Resources, Inc. June-
        August 2003.
        This information is based on a review of the references Mobley and Keller 1991, Thiel 1997,
        and Mobley 1987.

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GLOSSARY

absorption

In relation to health physics, process in which radiation energy is imparted to and radionuclides are transported to bodily fluids, tissues, and organs.

activation

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

albedo dosimeter

Thermoluminescent dosimeter that measures the thermal, intermediate, and fast neutrons scattered and moderated by the body or a phantom from an incident fast neutron flux.

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.

annual dose equivalent

Dose equivalent received in a year in units of rem or sievert. See *dose*.

anticontamination clothing (anti-Cs)

Special clothing worn by personnel for protection from radiological contamination.

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 (β) 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.

collective dose equivalent
Sum of the dose equivalents of all individuals in an exposed population in units of person-rem or person-sievert. See *dose*.

collective effective dose equivalent
Sum of the effective dose equivalents of all individuals in an exposed population in units of person-rem or person-sievert. See *dose*.

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
>   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 highly enriched uranium 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. Various terms narrow the type of dose, and some are additive:
>
>>   Absorbed dose is the amount of energy deposited in a substance by ionizing radiation.
>>
>>   Collective dose is the sum of the doses to a specific population.
>>
>>   Committed dose is the dose over time (usually 50 years for workers) to a specific organ or tissue from an intake of radioactive material.
>>
>>   Cumulative dose is the sum of all doses to the same portion of the body or to the whole body over time.
>>
>>   Deep dose is the dose at a 1-centimeter depth in tissue (1,000 milligrams per square centimeter).
>>
>>   Effective dose is the sum of the equivalent doses in the principal tissues and organs of the body, each weighted by a tissue weighting factor that accounts for the probabilities of fatal and nonfatal cancers according to severity and the average length of life lost due to an induced cancer. It indicates the biological effect of the radiation exposure in that tissue.
>>
>>   Equivalent dose or dose equivalent is the absorbed dose in a tissue or organ multiplied by a weighting factor for the particular type of radiation.
>>
>>   Organ dose is the dose to a specific organ.
>>
>>   Penetrating dose is that from higher energy photon (gamma and X-ray) radiation and neutron radiation that penetrates the outer layers of the skin. Nonpenetrating dose is that from beta and lower energy photon radiation.
>>
>>   Personal dose equivalent is the dose equivalent in soft tissue below a specified point on the body at a specified depth.

Shallow dose is the dose at a 0.07-millimeter depth in tissue (7 milligrams per square centimeter).

Skin dose is the dose to the skin.

Whole-body dose is the dose to the entire body excluding the contents of the gastrointestinal tract, urinary bladder, and gall bladder.

dose equivalent (H)
Product of absorbed dose in units of rem or sievert in tissue multiplied by a weighting factor and sometimes by other modifying factors to account for the potential for a biological effect from the absorbed dose. See *dose.*

dose of record
(1) Dose records that the U.S. Department of Energy provided to the National Institute for Occupational Safety and Health as part of each worker's file. (2) Individual recorded dose such as that on a dosimetry card or in a dosimetry database.

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 *albedo dosimeter*, *pocket ionization chamber*, and *thermoluminescent dosimeter.*

dosimetry
Measurement and calculation of internal and external radiation doses.

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.

gray (Gy)
 International System unit of absorbed radiation dose, which is the amount of energy from any type of ionizing radiation deposited in any medium; 1 Gy equals 1 joule per kilogram or 100 rads.

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 (HEU)
 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.

ionization chamber
　　Device that measures exposure or radiation dose.

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.

nuclear waste
> See *radioactive waste*.

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.

personal dose equivalent [$H_p(d)$]
> Dose equivalent in units of rem or sievert in soft tissue below a specified point on the body at an appropriate depth d. The depths selected for personal dosimetry are 0.07 millimeters (7 milligrams per square centimeter) and 10 millimeters (1,000 milligrams per square centimeter), respectively, for the skin (shallow) and whole-body (deep) doses. These are noted as $H_p(0.07)$ and $H_p(10)$, respectively. The International Commission on Radiological Measurement and Units recommended $H_p(d)$ in 1993 as dose quantity for radiological protection.

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 dosimeter
See *pocket ionization chamber*.

pocket ionization chamber (PIC)
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.

primary loop
Closed experimental system through which coolant flows as part of the control for a nuclear reaction using the main reactor as the primary source for neutron flux. See *secondary loop*.

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.

Quality factor, Q
Historical value assigned to reflect the average effectiveness of a particular kind of radiation in producing biological effects in humans, now called radiation weighting factor. The quality factor multiplied by the absorbed dose yields the dose equivalent. See *dose*.

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).

secondary loop
> System in a reactor that receives transfer heat from a primary system. The secondary system is not exposed to the reactor core and is free of radioactivity under normal circumstances. See *primary loop*.

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.

scram
> Sudden shutdown of the nuclear reaction in a reactor, typically caused by rapid insertion of control/safety rods, when a dangerous or undesirable condition occurs.

shallow dose equivalent [SDE, H_s, $H_p(0.07)$]
> Dose equivalent in units of rem or sievert at a depth of 0.07 millimeters (7 milligrams per square centimeter) in tissue equal to the sum of the penetrating and nonpenetrating doses.

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

sievert (Sv)
> International System unit for dose equivalent, which indicates the biological damage caused by radiation. The unit is the radiation value in gray (equal to 1 joule per kilogram) multiplied by a weighting factor for the type of radiation and a weighting factor for the tissue; 1 Sv equals 100 rem.

skin dose
> See *shallow dose equivalent*.

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

spent fuel storage basin
> Pool or pit of reinforced concrete filled with water for storage of spent nuclear fuel. The water is shielding and coolant.

thermoluminescence
> Property that causes a material to emit light as a result of heat.

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

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 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
> See *X-ray radiation*.

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.