ORAU TEAM
Dose Reconstruction
Project for NIOSH

Oak Ridge Associated Universities I Dade Moeller & Associates I MJW Corporation

Document Title:	Document Number:	ORAUT-TKBS-0007-2
Idaho National Engineering and Environmental Laboratory - Site Description	Revision:	02
	Effective Date:	07/29/2005
	Type of Document:	TBD
	Supersedes:	Revision 01

Subject Expert: Norman D. Rohrig

Document Owner
Approval: Signature on File — Approval Date: 07/18/2005
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☐ New ☐ Total Rewrite ☒ Revision ☐ Page Change

DOCUMENTS MARKED AS A TOTAL REWRITE, REVISION, OR PAGE CHANGE REPLACE THE PRIOR REVISION. PLEASE DISCARD / DESTROY ALL COPIES OF THE PRIOR REVISION.

PUBLICATION RECORD

EFFECTIVE DATE	REVISION NUMBER	DESCRIPTION
08/31/2003	00-A	New technical basis document for the Idaho National Engineering and Environmental Laboratory – site description. Initiated by Norman D. Rohrig.
10/02/2003	00-B	Incorporates NIOSH and internal review comments. Initiated by Norman D. Rohrig.
11/07/2003	00	First approved issue. Initiated by Norman D. Rohrig.
06/14/2004	01-A	Corrects dates of operation in Section 2.4.11 and adds OGC required paragraphs. Initiated by Norman D. Rohrig.
07/28/2004	01	Approved issue of Revision 01. Initiated by Norman D. Rohrig.
04/28/2005	02-A	Adds additional descriptive material on ICPP in response to union outreach comments. Initiated by Norman D. Rohrig.
05/25/2005	02-B	Incorporates internal review comments. Initiated by Norman D. Rohrig.
07/01/2005	02-C	Incorporates NIOSH review comments. Initiated by Norman D. Rohrig.
07/29/2005	02	Approved issue of Revision 02. Retraining is not required. Initiated by Norman D. Rohrig.

TABLE OF CONTENTS

Section Page

Publication Record 2

Acronyms and Abbreviations 7

2.1 Introduction 13

2.2 Test Area North 16
2.2.1 Technical Support Facilities 16
2.2.1.1 Hot Shop/Warm Shop/Hot Cells Annex, 1955 to present 16
2.2.1.2 TAN 607 Storage Pool 1955 to present 17
2.2.1.3 Storage Pads TAN 690 and TAN 691 17
2.2.1.4 Radioactive Liquid Waste Disposal System 1959 to present 18
2.2.1.5 Radioactive Parts Service and Storage Area 1955 to present 18
2.2.1.6 TAN 607 Radiography Facility 18
2.2.2 Initial Engine Test Facility, 1955 to 1966 19
2.2.2.1 Heat Transfer Reactor Experiment No.1, December 27, 1955 to January 3, 1959 19
2.2.2.2 Heat Transfer Reactor Experiment No. 2, July 1957 to March 28, 1961 20
2.2.2.3 Heat Transfer Experiment No. 3, 1958 to December 1960 20
2.2.2.4 Systems for Nuclear Auxiliary Power 10A Transient No.1, Early 1960s 20
2.2.2.5 Systems for Nuclear Auxiliary Power 10A Transient No. 3, April 1, 1964 21
2.2.2.6 Systems for Nuclear Auxiliary Power 10A Transient No. 2, 1965 to January 11, 1966 21
2.2.3 Water Reactor Research Test Facility, 1958 to 1973 21
2.2.3.1 Shield Test Pool Facility 22
2.2.3.2 Low Power Test Facility (LPTF), 1958 to 1973 22
2.2.3.2.1 Critical Experiment Tank, 1958 to 1960 23
2.2.3.2.2 Hot Critical Experiment, 1958 to March 28, 1961 23
2.2.3.2.3 Split Table Reactor System, 1971 23
2.2.3.2.4 Fast Spectrum Refractory Metals Reactor (710), March 1962 to 1968 23
2.2.3.2.5 Cavity Reactor Critical Experiment, May 17, 1967, to Early 1970s 23
2.2.3.2.6 Spherical Cavity Reactor Critical Facility, 1972 to November 1973 24
2.2.3.2.7 High Temperature Marine Propulsion Reactor (630-A), 1962 to 1964 24
2.2.4 Loss of Fluid Test Facility, 1973 to July 9, 1985 24
2.2.5 Specific Manufacturing Capability (SMC), 1985 to present 25

2.3 Idaho Nuclear Technology and Engineering Center, November 1951 to present 25
2.3.1 Fuel Processing Facility, CPP 601/602, February 1953 to 1992 26
2.3.2 Ancillary Facilities 28
2.3.3 INTEC Fuel Storage Facility (CPP-603), 1950 to present 29
2.3.4 High Level Liquid Waste Underground Storage Tanks (Tank Farm), 1951 to present 30
2.3.5 Waste Calcining Facility-1 and New Waste Calcining Facility, December 1963 To 2002 31

2.3.6 Fluorinel Dissolution Process and Fuel Storage Facility, 1984 to present 31
2.3.7 INEEL Comprehensive Environmental Response Compensation and Liability Act Disposal Facility Complex, July 2003 to present 32
2.3.8 TMI-2 Independent Spent Fuel Storage Installation – CPP-1774 to present 32

2.4 Argonne National Laboratories-West, February 1951 to present 33
2.4.1 Experimental Breeder Reactor No. I, April 1951 to December 30, 1963 33
2.4.2 Boiling Water Reactor Experiment No. 1, Late 1953 to July 22, 1954 33
2.4.3 Boiling Water Reactor Experiment No. 2, October 19, 1954, to March 1955 34
2.4.4 Boiling Water Reactor Experiment No. 3, June 9, 1955, to December 1956 34
2.4.5 Boiling Water Reactor Experiment No. 4, December 3, 1956, to June 1958 34
2.4.6 Boiling Water Reactor Experiment No. 5, February 9, 1962 to September 1964 35
2.4.7 Zero Power Reactor No. 3, October 1955 to November 1970 35
2.4.8 Argonne Fast Source Reactor, October 29, 1959, to Late 1970 36
2.4.9 Transient Reactor Test Facility, February 23, 1958, to April 1994 36
2.4.10 Experimental Breeder Reactor No. II, May 1961 to September 30, 1994 37
2.4.11 Hot Fuel Examination Facility, 1964 to Present 37
2.4.12 Zero Power Physics (Plutonium) Reactor, April 18, 1969 to April 1992 (standby) 38
2.4.13 Neutron Radiography Facility, October 1, 1977 to present 38
2.4.14 Fuel Assembly and Storage Building, 1970 to present 39
2.4.15 Other ANL-West Support Facilities 39

2.5 Radioactive Waste Management Complex, May 1952 to present 40
2.5.1 Subsurface Disposal Area 40
2.5.2 Intermediate Level Transuranic Storage Facility 40
2.5.3 Transuranic Storage Area 41
2.5.4 Stored Waste Examination Pilot Plant 41
2.5.5 Transuranic Package Transporter Loading Station 41
2.5.6 Advanced Mixed Waste Treatment Project 41

2.6 Central Facilities Area, August 2, 1943, to present 41
2.6.1 The Hot Laundry 42
2.6.2 The Health Physics Instrument Laboratory 42
2.6.3 DOE Laboratory Accreditation Procedure Irradiation Facility 42
2.6.4 Radiological Environmental Science Laboratory 43
2.6.5 CF-674 Building 43

2.7 Test Reactor Area, MARCH 31, 1952, to present 43
2.7.1 Materials Test Reactor, March 31, 1952, to April 23, 1970 43
2.7.2 Engineering Test Reactor, September 19, 1957, to December 1981 44
2.7.3 Advanced Test Reactor, July 7, 1967, to present 45
2.7.4 Reactivity Measurement Facility, February 11, 1954, to April 10, 1962 45
2.7.5 Advanced Reactivity Measurement Facility No.1, October 10, 1960, to 1974 45
2.7.6 Advanced Reactivity Measurement Facility No. 2, December 14, 1962 to 1968 45
2.7.7 Coupled Fast Reactivity Measurement Facility, 1968 to 1991 46
2.7.8 Engineering Test Reactor Critical Facility, May 20, 1957, to 1982 46
2.7.9 Advanced Test Reactor Critical Facility, May 19, 1964 to present 46
2.7.10 TRA Hot Cell Facility 1954 to present 47

2.7.11 TRA Gamma Facility, 1955 to unknown 47
2.7.12 Radiation Measurements Laboratory, 1952 to present 47
2.7.13 Radiochemistry Laboratory 48
2.7.14 Liquid Waste Disposal Ponds 48
2.7.15 High-Level Liquid Waste Tanks and Transfer Facility 48

2.8 Auxiliary Reactor Area, April 1958 to late 1990s 49
2.8.1 ARA-I 49
2.8.2 ARA-II/SL-1, February 23, 1960, to April 6, 1961 49
2.8.3 ARA-III 50
2.8.4 ARA-IV, March 30, 1961 to May 29, 1964 50

2.9 Waste Reduction Operations Complex/Power Burst Facility/Special Power Excursion Reactor Test Areas 51
2.9.1 SPERT-I, June 11, 1955 to 1964 51
2.9.2 SPERT-II, March 11, 1960, to October 1964 51
2.9.3 SPERT-III, December 19, 1958, to June 1968 51
2.9.4 SPERT-IV, July 24, 1962, to August 1970 52
2.9.5 Power Burst Facility, September 22, 1972, to 1985 52
2.9.6 Lead Storage Facility 52
2.9.7 Waste Experimental Reduction Facility, 1982 to 2001 53
2.9.8 Mixed Waste Storage Facility 53

2.10 Organic Moderated Reactor Experiment, September 17, 1957, to April 1963 53

2.11 Test Grid III, 1957 to April 24, 1970 54
2.11.1 Fuel Element Burn Tests 54
2.11.2 Fission Products Field Release Tests, July 25, 1958, to September 26, 1958 54
2.11.3 Relative Diffusion Tests, November 30, 1967, to October 1, 1969 55
2.11.4 Experimental Cloud Exposure Study, May 3, 1968, to April 24, 1970 55

2.12 Experimental Dairy Farm/Experimental Field Station 55
2.12.1 Controlled Environmental Radioiodine (Release) Tests, May 27, 1963, to December 1977 55

2.13 INEEL Research Center, 1984 to present 56

2.14 Army Reentry Vehicle Facility Site (or Station), 1965 to May 1996 56

References 58

Glossary 63

LIST OF TABLES

Table Page
2-1 INEEL prime site contractors 14
2-2 Other operating facility technical area contractors 14
2-3 ICPP 601/602 process cell information 57

LIST OF FIGURES

Figure Page
2-1 Map of the Idaho National Engineering and Environmental Laboratory 15

ACRONYMS AND ABBREVIATIONS

a	alpha particle
AEC	Atomic Energy Commission
AECL	Administrative Exposure Control Level
AEDE	annual effective dose equivalent
AFSR	Argonne Fast Source Reactor
AL	Analytical Laboratory
AMWTP	Advanced Mixed Waste Treatment Program
ANC	Aerojet Nuclear Corporation
ANL	Argonne National Laboratory
ANL-E	Argonne National Laboratory East
ANL-W	Argonne National Laboratory West
ANP	Aircraft Nuclear Propulsion
ANPP	Aircraft Nuclear Propulsion Program
APS	Atmospheric Protection System
ARA	Army Reactor Area (later Auxiliary Reactor Area)
AREA	Army Reactor Experimental Area
ARMF-1	Advanced Reactivity Measurements Facility No.1
ARMF-2	Advanced Reactivity Measurements Facility No.2
ARVFS	Army Reentry Vehicle Facility Site
ATR	Advanced Test Reactor
ATRC	Advanced Test Reactor Critical Facility
ß	beta particle
BNFL	British Nuclear Fuels Limited
BORAX	Boiling Water Reactor Experiment
BORAX I	Boiling Water Reactor Experiment No. 1
BORAX II	Boiling Water Reactor Experiment No. 2
BORAX III	Boiling Water Reactor Experiment No. 3
BORAX IV	Boiling Water Reactor Experiment No. 4
BORAX V	Boiling Water Reactor Experiment No. 5
BWR	Boiling Water Reactor
C	Celsius/centigrade
CA	Contamination Area
CAM	continuous air monitors
CAMU	Corrective Action Management Unit
CAS	Criticality Accident System
CDC	Capsule Driver Core
CE	Critical Experiment
CEL	Chemical Engineering Laboratory
CEDE	committed effective dose equivalent
CERCLA	Comprehensive Environmental Response Compensation and Liability Act
CERT	Controlled Environmental Radioiodine Tests (later Radionuclides)
CET	Critical Experiment Tank
CFA	Central Facilities Area
CFRMF	Coupled Fast Reactivity Measurement Facility
c/m	counts per minute
CPP	Chemical Processing Plant
CRCE	Cavity Reactor Critical Experiment

CTF	Core Test Facility
D&D	decontamination and decommissioning
DAC	Derived Air Concentration
DE	dose equivalent
DL	detection limit
DOD	U.S. Department of Defense
DOE	U.S. Department of Energy
DU	depleted uranium
DOE-ID	DOE-Idaho Operations Office
DOELAP	DOE Laboratory Accreditation Program
EBR	Experimental Breeder Reactor
EBR-I	Experimental Breeder Reactor No.1
EBR-II	Experimental Breeder Reactor No.2
EBOR	Experimental Beryllium Oxide Reactor
EBWR	Experimental Boiling Water Reactor
ECCS	Emergency Core Cooling System
EEOICPA	Energy Employees Occupational Illness Compensation Program Act
EFS	Experimental Field Station
EOCR	Experimental Organic Cooled Reactor
EPA	Environmental Protection Agency
ERDA	Energy Research and Development Administration
ETF	Effluent Treatment Facility
ETR	Engineering Test Reactor
ETRC	Engineering Test Reactor Critical
EXCES	Experimental Cloud Exposure Study
F	Fahrenheit
FARET	Fast Reactor Test
FASB	Fuel Assembly and Storage Building
FAST	Fluorinel Dissolution Process and Fuel Storage Facility
FCF	Fuel Cutting Facility
FDF	Fluorinel Dissolution Facility
FEBT	Fuel Element Burn Tests
FFTF	Fast-Flux Test Facility
FMF	Fuel Manufacturing Facility
FPF	Fuel Processing Facility
FPR	Fuel Processing Restoration
FPFRT	Fission Product Field Release Test
FSF	Fuel Storage Facility
?	gamma
GCRE	Gas-Cooled Reactor Experiment
GE	General Electric Corporation
HCA	High Contamination Area
HEPA	high-efficiency particulate air
HEU	highly enriched uranium
HFEF	Hot Fuel Examination Facility
HLLW	high-level liquid waste

HLW	high-level waste
HOTCE	Hot Critical Experiment
HP	health physicist/health physics
HPIL	Health Physics Instrument Laboratory
HPP	Hot Pilot Plant
HRA	High Radiation Area
HSL	Health Services Laboratory
HTRE	Heat Transfer Reactor Experiment
IBM	International Business Machines
IC	Initial Criticality
ICP	Idaho Cleanup Project
ICDF	INEEL CERCLA Disposal Facility
ICPP	Idaho Chemical Processing Plant
ICRP	International Commission on Radiological Protection
ID	Idaho
IDO	Idaho Operations Office
IWP	Industrial Waste Pond
IFSF	Irradiated Fuel Storage Facility
IET	Initial Engine Test
IFR	Integral Fast Reactor
ILTSF	Intermediate-Level Transuranic Storage Facility
IMBA	Internal Modular Bioassay Analysis
INC	Idaho Nuclear Corporation
INL	Idaho National Laboratory
INEC	Idaho Nuclear Energy Commission
INEL	Idaho National Engineering Laboratory
INEEL	Idaho National Engineering and Environmental Laboratory
INTEC	Idaho Nuclear Technology and Engineering Complex
IRC	INEEL Research Center
IREP	Interactive RadioEpidemiological Program
ISU	Idaho State University
ISFSI	Independent Spent Fuel Storage Installation
IWP	Industrial Waste Pond
keV	Kiloelecton volt
kVp	Kilovolt peak
kW	kilowatt
LANL	Los Alamos National Laboratory
LLD	lower limit of detection
LMFBR	Liquid Metal Fast Breeder Reactor
LOCA	Loss-of-Coolant Accident
LOFT	Loss-of-Fluid Test
LPTF	Low-Power Test Facility
LSC	Liquid Scintillation Counter
MDF	Materials Development Facility
MeV	megaelectron volt
ML-1	Mobile Low-Power Reactor

MPBB	Maximum Permissible Body Burden
MPLB	Maximum Permissible Lung Burden
MPOB	Maximum Permissible Organ Burden
mR	milliroentgen
mrem	millirem
ms	millisecond
MTR	Materials Testing Reactor
MW	megawatt
NaK	Sodium potassium
NASA	National Aeronautics and Space Administration
NCRP	National Council on Radiation Protection and Measurements
NERP	National Environmental Research Park
N_f	neutron flux-fast
NIOSH	National Institute for Occupational Safety and Health
NOAA	National Oceanic and Atmospheric Administration
NOCTS	NIOSH OCAS Claims Tracking System
NPR	New Production Reactor
NRAD	Neutron Radiography Facility
NRC	Nuclear Regulatory Commission
NRF	Naval Reactors Facility
NRTS	National Reactor Testing Station
NTA	Nuclear Track Emulsion
N_{th}	neutron flux-thermal
NWCF	New Waste Calcining Facility
OCAS	(NIOSH) Office of Compensation Analysis and Support
OMRE	Organic Moderated Reactor Experiment
ORAU	Oak Ridge Associated Universities
PA	Posterior-Anterior (medical X-ray)
PAS	Personal Air Sampler
PBF	Power Burst Facility
PCB	polychlorinated biphenyl
PIC	Pocket Ionization Chamber (i.e., "pencil" dosimeter)
PIF	Process Improvement Facilities
psi	pounds per square inch
PWR	pressurized-water reactor
R	Roentgen
rad	radiation absorbed dose
RAF	Remote Analytical Facility
RAL	Remote Analytical Laboratory
RaLa	radioactive lanthanum
RAM	Radiation/Remote Area Monitor
RBE	Relative Biological Effectiveness
RBOF	Receiving Basin for Offsite Fuel
RCRA	Resource Recovery and Conservation Act
RCT	Radiological Control Technician
RDT	Relative Diffusion Test
rem	Roentgen equivalent man

RESL	Radiological Environmental Sciences Laboratory
rep	Roentgen equivalent physical
RHA	Radiation Hazards Analysis
RLWTF	Radioactive Liquid Waste Treatment Facility
RMF	Reactivity Measurement Facility
RML	Radiation Measurements Laboratory
RPSSA	Radioactive Parts Service and Storage Area
RU	recycled uranium
RWMC	Radioactive Waste Management Complex
RWP	Radiological Work Permit
RSAC	Radiological Safety Analysis Computer
RSWF	Radioactive Scrap and Waste Facility
SCRCE	Spherical Cavity Reactor Critical Experiment
SDA	Subsurface Disposal Area
SEC	Special Exposure Cohort
SID	Source to Image Distance
SIS	Special Isotope Separations
SL-1	Stationary Low-Power Reactor
SM-1	Stationary Medium-Power Reactor
SMC	Specific Manufacturing Capability
SNAP	Systems for Nuclear Auxiliary Power
SNAPTRAN	Systems for Nuclear Auxiliary Power Transient
SPERT	Special-Power Excursion Reactor Test
SPF	Sodium Processing Facility
SRS	Savannah River Site
SSD	Source to Skin Distance
SSSTF	Staging, Storage, Sizing, and Treatment Facility
STM	Stack Tritium Monitor
STEP	Safety Test Engineering Program
STP	Sewage Treatment Ponds
STPF	Shield Test Pool Facility
STR	split-table reactor
SWDF	Solid Waste Disposal System
SWEPP	Stored Waste Examination Pilot Plant
SWP	Safe Work Permit
t	thermal
?½	Half life of a nuclide or isotope
TAN	Test Area North
TBD	Technical Basis Document
TEPC	Tissue Equivalent Proportional Counter
TLD	Thermoluminescent dosimeter
TLND	Thermoluminescent neutron dosimeter
TMI	Three Mile Island
TRA	Test Reactor Area
TREAT	Transient Reactor Test
TRIGA	Training Research and Isotope General Atomic
TSA	Transuranic Storage Area
TRU	transuranics
TRUPACT	Transuranic Packaging Transporter

TSF	Technical Support Facility
USGS	U. S. Geological Survey
WAC	Waste Acceptance Criteria
WBRR	Western Beam Research Reactor
WCA	Waste Characterization Area
WCF	Waste Calcining Facility
WERF	Waste Experimental Reduction Facility
WIPP	Waste Isolation Pilot Plant
WROC	Waste Reduction Operations Complex
WRRTF	Water Reactor Research Test Facility
ZPR	Zero Power Reactor
ZPR III	Zero Power Reactor No.3
ZPPR	Zero Power Plutonium Reactor (later Zero Power Physics Reactor)

2.1 INTRODUCTION

The purpose of this document is to provide an Idaho National Engineering and Environmental Laboratory (INEEL) "Site" profile that contains technical basis information, to evaluate the total individual occupational dose for claimants under the Energy Employees Occupational Illness Compensation Program Act of 2000 (EEOICPA).

Technical Basis Documents and Site Profile Documents are general working documents that provide guidance concerning 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 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 facility" as defined in the EEOICPA (42 U.S.C. § 7384l (5) and (12)).

The INEEL 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 Technical Basis Document (TBD) is to assist in the evaluation of worker dose from INEEL processes using the methodology in NIOSH OCAS-*IG-001 External Dose Reconstruction Implementation Guideline (NIOSH 2002a)* and *OCAS-IG-002 Internal Dose Reconstruction Implementation Guideline (NIOSH 2002b)*.

This TBD provides supporting technical data with claimant-favorable assumptions to evaluate the total INEEL occupational dose that can be reasonably associated with worker radiation exposure as covered under EEOICPA legislation. The documentation in this TBD addresses evaluation of monitored and unmonitored worker exposure and missed dose. In addition, it presents the technical basis of methods used to prepare the INEEL worker dose records for input to the NIOSH Interactive RadioEpidemiological Program (IREP) and the Internal Modular Bioassay Analysis (IMBA) computer codes used to evaluate worker dose. It also presents information on the uncertainty for recorded INEEL 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 may be unavailable.

Proving the Principle, *A History of the Idaho National Engineering and Environmental Laboratory, 1949-1999*, by Susan M. Stacy (Stacy 2000) was used as a resource in the development of facility and process descriptions.

INEEL or "the Site" is an 890-square-mile reservation encompassing almost 572,000 acres with a maximum distance of about 39 miles from north to south and 36 miles from east to west. It is 30 to 60 miles 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, the NRTS became the Idaho National Engineering Laboratory (INEL) and, in 1997, the Idaho National Engineering and Environmental

Laboratory (INEEL). On February 1, 2005, the site became the Idaho National Laboratory (INL) combining the research side of the INEEL and ANL-W and the Idaho Cleanup Project (ICP) working on closure of inactive portions of the site.

INEEL is unique among U.S. Department of Energy (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 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 dates contractors other than the prime contractor operated them.

Table 2-1. INEEL prime site contractors.

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.

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 (ANPP)	
1952–1961	General Electric
Specific Manufacturing Capability	
1983–1986	Exxon Nuclear Idaho Company
1986–1991	Rockwell INEL
1991–1994	Babcock & Wilcox Idaho, Inc.

Dosimetry services at INEEL were unique among DOE facilities in that DOE Idaho Operations Office (DOE-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 also provided portable radiation survey instruments and maintained and calibrated them. Originally, DOE-ID personnel at the Radiological Environmental Sciences Laboratory (RESL) in Building CF-690 in Central Facilities Area maintained exposure histories of personnel based on dosimetry records, including bioassay data. With the advent of the DOE Laboratory Accreditation Program (DOELAP), dosimetry responsibility was 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.

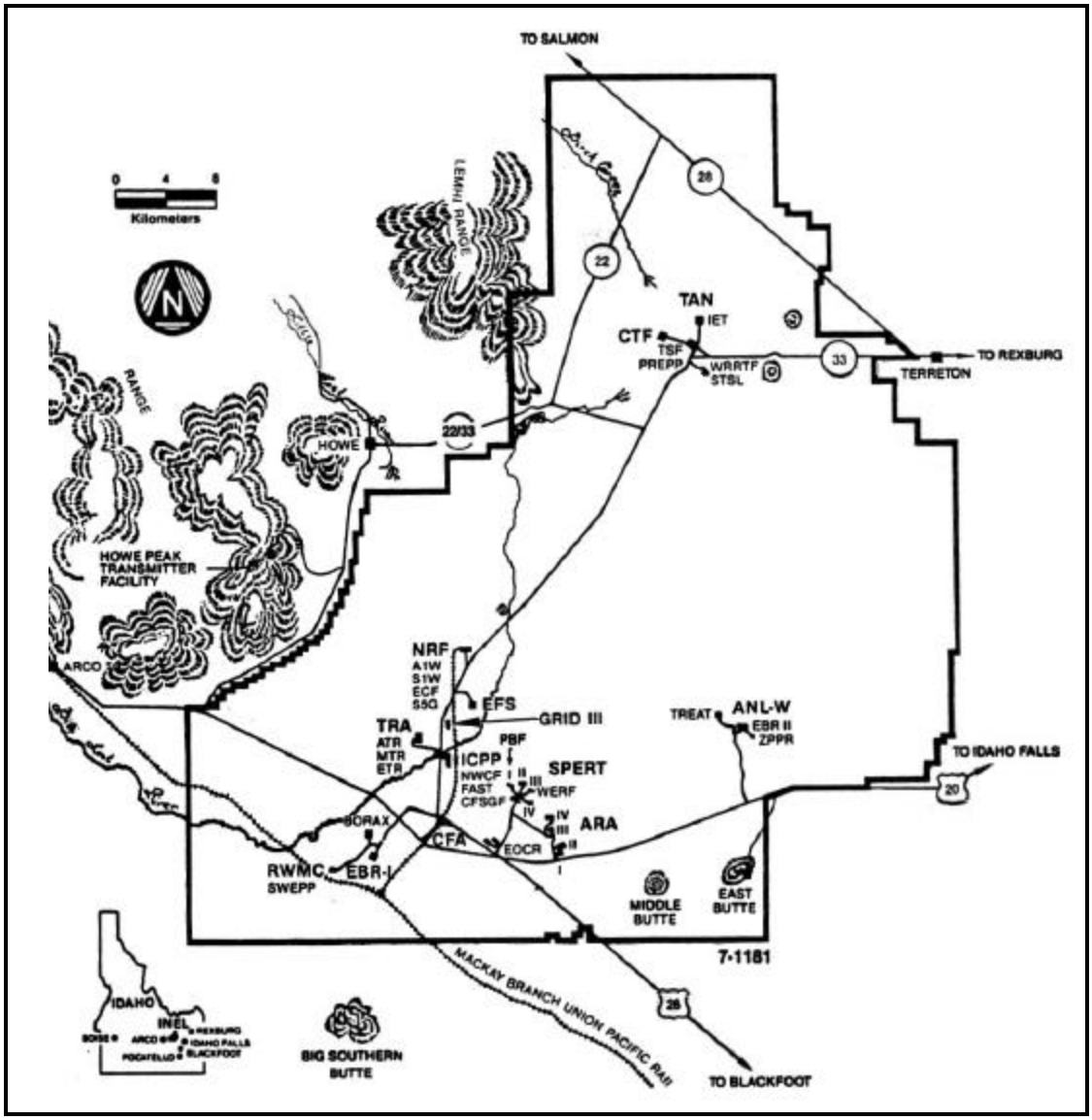

Figure 2-1. Map of the Idaho National Engineering and Environmental Laboratory. Facilities include the Argonne National Laboratory-West (ANL-W), where the Experimental Breeder Reactor No. 2 (EBR-II), Transient Reactor Test Facility (TREAT) and Zero Power Plutonium Reactor (ZPPR) are located; Auxiliary Reactor Area (ARA); Central Facilities Area (CFA); Idaho Chemical Processing Plant (ICPP) [now the Idaho Nuclear Technology and Engineering Complex (INTEC)]; Experimental Breeder Reactor No. 1 (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 (FEBT) A and B occurred; Test Reactor Area (TRA); the Experimental Field Station (EFS); Naval Reactors Facility; Test Area North (TAN), where the Initial Engine Tests (IETs) occurred; and the Core Test Facility (CTF) at TAN.

Personnel working at INEEL in designated radiological areas were typically required to wear state-of-the-art dosimetry (film badges, thermoluminescent dosimeters [TLDs], personal ion chambers [PICs], respiratory protection, anti-contamination clothing, etc.). Facilities and radiological areas were monitored by remote and portable radiation/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. In cases where airborne radioactivity might be present or where internal exposure was possible, applicable respirators were provided to prevent/reduce internal exposure.

Engineered systems were incorporated as practicable to minimize the potential for radiological 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 within Site locations.

In addition to the nuclear reactor experiments discussed below in relation to the TRA, there have been other such experiments at INEEL. Table 2-3 lists common radionuclides for reactors.

2.2 TEST AREA NORTH

TAN is 30 miles northeast of the Central Facilities Area. General Electric Corporation (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 Initial Engine Test (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 the TAN area next to the aircraft hanger built to support the ANP Program. LOFT was a smaller version of a commercial pressurized-water reactor designed to allow, create, or recreate 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 Facilities

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 present

The TAN Hot Shop is in the TAN-607 building. It is 51 feet wide by 165 feet long by 55 feet high. The walls are 7 feet thick and the windows are 6 feet 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 feet wide by 80 feet long by 50 feet 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 feet wide by 35 feet 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^{-1} 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.

Internal exposure potential exists from airborne radioactivity of mixed fission products and activation products associated with reactor operation (Table 2.1.3).

External exposure exists from mixed fission products and mixed activation products (MAP). Radiation levels varied from background levels to measured levels greater than 50 rad hr^{-1}, principally gamma rays with energy greater than 250 keV.

2.2.1.2 TAN 607 Storage Pool 1955 to present

The TAN Storage Pool is adjacent to the north side of the Hot Shop. It is 70 feet long, 48 feet wide, and 24 feet deep. An underground passageway (19 feet by 24 feet) 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 feet long, 24 feet wide and 24 feet deep. The top of the passageway under the shield wall is 5 feet 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 (MFP) products, activation products and transuranics. 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^{-1} (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 690 and TAN 691

The Storage Pads are near the main east-west railroad west of the Hot Shop and east of the turntable. The TAN 690 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^{-1} of transuranics. External radiation levels are less than or equal to 0.1 mrem hr^{-1}.

Storage Pad TAN-691 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 Power 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^{-1} gamma and about 40 mrem hr^{-1} neutron. The casks are periodically surveyed and assessed for radioactive contamination.

Internal exposure potential exists from potential airborne leaks.

External exposure exists from activities proximate to the casks. Not all personnel wear neutron dosimetry.

2.2.1.4 Radioactive Liquid Waste Disposal System 1959 to present

A radioactive liquid waste system (TAN 666), which was built in 1959, collected and processed intermediate-level liquid wastes generated in the TAN area and transferred them to one of three underground 10,000-gallon, 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-gallon underground liquid waste feed tanks (V9 and V10). Solids were separated and sent to the RWMC. Originally, the liquid effluent was combined with low-level radioactive liquid waste and discharged to a disposal well. In 1972, INEEL 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 gallons per year (ERDA 1977). From 1959 to 1974, TAN reported liquid effluent releases to the disposal well or pond of 58 curies, with highest releases in 1959, 1968, and 1969.

Internal exposure exists from potential airborne mixed fission products and activation products.

External exposure exists from activities around the liquid waste system.

2.2.1.5 Radioactive Parts Service and Storage Area 1955 to present

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 Heat Transfer Reactor HTRE-2 and HTRE-3 reactors. The buildings and the 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^{-1} beta near contact ($^{90}Sr/Y$). The gates and buildings are locked when access is not required.

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 of leaking packages or disturbing contaminated soil areas containing mixed fission products and/or activation products.

External exposure comes from several items in storage that have had radiation levels in the R hr^{-1} range from mixed fission products and activation products.

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-curie range when first purchased. 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.

External exposure occurred from activities associated with radiographic sources.

2.2.2 Initial Engine Test Facility, 1955 to 1966

The IET (TAN-620) was approximately 1 mile 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 feet thick and ceilings 3 feet thick. The floor of the facility was 15 feet below grade with 14 feet 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. 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) Transient (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 various 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, Rothstein, and Culver 1962).

Internal exposure was possible from airborne mixed fission products and activation products. It was minimized by a filtered intake air system that kept the air pressure in the occupied areas positive in relation to outside air. The exhaust from the HTRE jet engines was vented up a 150-foot stack.

External exposure occurred from the activation and mixed fission product inventory after tests were completed and personnel worked on the associated test assemblies.

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 hours at full power, 20 MW thermal, exceeding the design requirement of 100 hours. During the first 6 hours 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, Rothstein, and Culver 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 inches 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 requiring removal of the core from the shield. This converted it to a materials test reactor, which subjected test fuels to environments reaching 2,800°F for extended periods and higher temperatures for short periods. The ANP Program advanced the technology of high-heat ceramic reactor fuels. HTRE-2 operated for 1,299 hours at powers up to 14 MW. Test sections consisted of metallic fuel elements combined with air-cooled zirconium hydride moderators and beryllium oxide fuel elements for use in ceramic reactors (Thornton, Rothstein, and Culver 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 hours, 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 2000°F without the help of any chemical fuel (Thornton, Rothstein, and Culver 1962).

On November 18, 1958, a nuclear excursion occurred during the IET No. 13 test. 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 boundaries of the NRTS. The maximum dose rate observed in the Assembly and Maintenance area and approximately 3,000 feet from the cloud centerline was .04 mrem hr^{-1}. The maximum fallout observed, at about 4 hours after the incident, measured 0.8 to 2.0 mrem hr^{-1} at contact roughly 1 1/2 miles 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. After the HTRE program ended, HTRE 2 and 3 reactors were parked in the RPSSA. The HTRE engines were moved to the EBR-1 historic site in 1988.

<u>Internal exposure</u> 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 mixed fission products 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 (NCRP) methodology (NCRP 1996); tests 10 and 4 were ranked the highest. Seven tests were ranked group 2. Fission Product inventories were based on documented reactor operating histories (RAC 2002). Operations and support personnel were inside the pressurized control room (TAN-620) during reactor operation during the testing phase of the HTRE reactor program minimizing their exposure potential.

<u>External Exposure</u> occurred from the mixed fission product and activation product inventory during work associated with test assemblies after tests were completed.

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 SNAP10A/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. The SNAP 10A/2 reactors were 9 inches in diameter by 12 inches long and were composed of a sodium/potassium (NaK)-cooled core containing 37 rods of fully enriched uranium in a zirconium-hydride matrix.

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 (SNAP10A/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 instead of 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 milliseconds (ms). More than 99% of the fission products 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. 1967).

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 SNAP10A/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 4681 curies 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 hours after the tests. Wind had to be southwest (180° to 240°) between 10 and 30 mph (Cordes et al.1965).

<u>Internal exposure</u> potential during the SNAP 10A test series was well controlled by requiring personnel to stay inside the IET 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.

<u>External exposure</u> to mixed fission products and mixed activation products 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.

2.2.3 <u>Water Reactor Research Test Facility, 1958 to 1973</u>

The WRRTF is approximately 1.25 miles southeast of the TSF area. The first facility constructed at WRRTF was the Low Power Test Facility (LPTF) in 1958. This facility was also known as Semiscale

and 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.

The Semiscale facility in LPTF was a forerunner of the LOFT. It was a scaled mockup of one loop of a four-loop pressurized-water reactor (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.

Due to the electrically heated experiment cores in the facilities listed in the subsections of 2.2.3, internal and external exposure from neutron reactions in the reactors did not exist. 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 feet 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 the NRTS (Walsh 1961). The facility would later become known as EBOR. Since 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-curie ^{137}Cs sources in a radiography-type environment in which the source material was cranked out of a shielding cask to be in the proximity of the piping. The Cs and tracer sources have been removed.

<u>Internal exposure</u> at the STPF was not known to occur due to the low power of Susie, the water environment, and use of sealed small sources.

<u>External exposures</u> occurred from the use of sealed sources, primarily ^{137}Cs.

2.2.3.2 Low Power Test Facility (LPTF), 1958 to 1973

The LPTF was used to conduct several low-power (less than 100-watt) 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-foot-thick wall between them. The walls between the cells and the control room are 5 feet thick for a height of 30 feet. The outside wall of the IC cell is 2 feet thick, and the outside wall of the CE cell is 3 feet thick for a height of 30 feet (Kunze and Chase 1970).

The construction of the facility was such that more than one reactor program could be running at the same time. 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 inch 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 watt for a period of 1 to 3 hours. The reactor could be operated at 100 watts 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 was operated in the CE cell of the LPTF.

The reactor was a split-table type, 7 ft 2 in wide and 11 ft long. The reactor was opened and closed using a hydraulic system. 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 watts thermal not to exceed 1000 watts or 10 kilowatt hours 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.

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.

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.

Internal exposure was not known to occur.

External exposures occurred during cell entry after a reactor run.

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 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.

The LOFT facility was built in a steel domed reactor containment vessel, 97 feet 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 LOFT 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-month period, 8,780 curies of noble gas (^{88}Kr) and 0.09 curie of Iodine (^{131}I) were released to the environment (Hoff, Chew, and Rope 1986; Stachew 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^{-1} beta gamma. The short-lived fission products would die off rapidly, reducing the general fields to = 10 mrad hr^{-1} beta gamma.

2.2.5 Specific Manufacturing Capability (SMC), 1985 to present

The SMC project is in the ANP Program aircraft hanger (TAN 629) and surrounding buildings. The project consists of Phase I, Phase II, and support facilities. A Materials Development Facility (MDF), located in TAN 607A and the TAN Hazardous Waste Storage Area, located in TAN-628, were formerly 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 for the Army made out of depleted uranium (DU). The major radioactivity in the DU is ^{238}U, ^{234}Th, ^{234m}Pa, and ^{234}U.

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 and 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.

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

2.3 IDAHO NUCLEAR TECHNOLOGY AND ENGINEERING CENTER, November 1951 to present

INTEC, formerly the Idaho Chemical Processing Plant (ICPP or CPP), commonly known as the "Chem Plant", is 53 miles west of Idaho Falls and occupies 200 acres in the middle of the INEEL 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 INEEL, 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 granular solid form.

In addition to calcination and fuel reprocessing operations, INTEC is a major fuel storage facility (FSF). INTEC houses a HLLW evaporation facility, HLLW storage (Tank Farm), airborne radioactive waste processing, non-radioactive 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 safely treat high-level and liquid radioactive waste that resulted from reprocessing spent fuel continues, past environmental releases are being remediated, and some facilities have been decontaminated and decommissioned. Inactive INTEC facilities are being evaluated for D&D.

Table 2-3 lists the primary long-lived nuclides in the INTEC processes.

Internal exposure might have occurred when workers were near breached and or leaking systems containing mixed fission products, transuranics, activation products, etc.

External exposure occurred during maintenance work, laboratory work, fuel cutting, and other support work which briefly exposed workers to radiation rates from background levels to measured levels of = 50 rad hr^{-1} beta/gamma.

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 transuranics. Solvent extraction with hexone and tributyl phosphate (Boardman ~1956) separated uranium from the fission products.

The Process Building CPP 601 contains 29 heavily shielded underground process cells. The building is 250 feet long by 102 feet wide and extends up to 57 feet below grade and 38 feet 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 field office in V cell, cells were accessible only during shutdown periods for maintenance and decontamination activities. The process building was designed for handling modest quantities (up to several kg) of enriched spent fuel due to criticality considerations. It was designed to be remotely decontaminated so that hands on maintenance could occur. This design feature results in many activities being conducted with significant dose rates and/or contamination levels.

The Laboratory Building CPP 602 shares a common wall with CPP 601 and is 147 feet long, 102 feet wide, and about 80 feet 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 Y-12 in 10 liter polyethylene bottles in concentrations exceeding 250 grams uranium/liter (Lewis et al. 2000). Beginning in 1971 a denitrator in a glove box 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 Testing Reactor (MTR) fuel was dissolved in acid, the dissolving process liberated gases, one of which was 8 day half-life ^{131}I. The RaLa process recovered ^{140}Ba for its ^{140}La daughter product and shipped it to Los Alamos National Laboratory (LANL) for use in weapon research projects.

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 for use as targets in making Pu-238 in this demonstration project. The processing was done in CPP-601 and bottling of the product was done in the multi-curie cell of CPP-627. Afterwards the process did not separate the neptunium so it remained a minor constituent of the product and waste.

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 preserves 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 liters of the solution to a 5,000-gallon tank containing about 600 to 900 liters of water. The resulting criticality lasted about 20 minutes. No workers were exposed to gamma or neutron radiation from the criticality because the criticality occurred in an unoccupied below-ground 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 liters of solution (76 liters remained in the vessel) and unexplainably moved about 600 liters into a companion vessel. High radiation level (>25 R/hr) was discovered near the RaLa area above the waste tank. Fields beyond the guardhouse were 2 R/hr. The high fields were probably due to the large activity in the dissolved spent fuel outside of 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).

- Criticality Accident of January 25, 1961. About 40 liters of uranyl nitrate solution (200 grams of uranium per liter) was forced upward from a 5-inch-diameter section of an evaporator into a 24-inch-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 a pump. 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.

- ICPP Criticality Accident of October 17, 1978. A criticality event occurred in the first-cycle tributylphosphate extraction system in the CPP-601 process building at the ICPP. 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 (APS) at INTEC, which became operational in 1975, significantly reduced particulate emissions, and filtered all releases associated with the criticality event.

Internal exposure 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.

External exposure occurred from work performed in Radiation Areas (RAs) and High Radiation Areas (HRAs) containing the nuclides referenced in Table 2-3. Brief exposures to measured levels of 50 rad hr^{-1} existed in the cells during piping/valve changes, decontamination, maintenance activities, working production samples in the laboratories, etc.

2.3.2 Ancillary Facilities

New fuels presented special problems in reprocessing. The Process Improvement Facilities (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 also 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 curie of radioactive materials from Table 2-3 per laboratory. The waste stream discharged to the HPP waste system.

The HPP(CPP 640) was used to test unproven equipment and systems. The facility consisted of five cells and associated utilities. The partition between two of these 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 months in CPP 640 using two stages of burning to reduce the carbon content (Knecht et al. 1997). The ash was leached with a nitric/hydrofluoric acid mixture, extracted through three cycles of extraction and then converted to UO_3. Over 100 kg of uranium was 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 (RAF) in CPP-627 houses the Remote Analytical Laboratories (RAL), the multi-curie cell, a radio-chemistry laboratory, and a decontamination facility.

The decontamination facility in CPP-627 provided support for cleaning tools and equipment for INTEC and other INEEL 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^{-1} open window beta gamma were experienced for brief periods in the decontamination facility. Shielding other than temporary was not provided between the several work stations there.

The RAL provides two rows of 15 analytical boxes behind a 9" 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 INEEL 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^{-1}. During the analytical process the samples were diluted to reduce personnel exposure.

The multi-curie cell has walls 5' thick of barytes concrete and is designed to reduce the field from 75,000 Ci of 1.6 MeV gamma emitter to 1 mR/hr. There is also a radiochemistry laboratory nearby to support operations in the multi-curie cell. A walk-in hood in this area contained the custom processing facility. On 9 February 1991 a small explosion destroyed the 6" OD borosilicate glass dissolver section, contaminating 4 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 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 present

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

Levels of airborne radioactivity of MFP around the 603 unlined storage pools were a chronic problem from sodium contaminated EBR 1 fuel which also led to contamination in the building. Efforts to clean up the water were aggravated by deionized water attacking the concrete pool. Anti-contamination 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 RCG_a 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 (FCF). 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 Savannah River Site (SRS) reactors. The 14-foot elements were clad in aluminum and had to be cut to 18-inch lengths to fit in the dissolver vessel. Irradiation of the fuel changed metallurgical characteristics, so instead of 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 decontamination and decommissioning (D&D).

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 mixed fission products from the pool water.

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.

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

The HLLW Tank Farm includes 11 underground stainless-steel, 300,000-gallon 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 should a problem develop 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 INEEL is composed 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 WINCO ALARA committee. The workers wore TLDs and electronic dosimeters and/or self-reading pencil dosimeters. Although this dose was separately kept track of, it is also included in the cumulative dose records for those employees.

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 groundwater contamination have resulted from these operations.

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

Internal exposure potential existed from airborne radioactivity created during flushing operations, valve repairs, or other maintenance activities in contamination areas from mixed fission products and activation products.

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^{-1}.

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 Argonne National Laboratory (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-gallon 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 feet of solid waste. This was a reduction in volume of more than 9 to 1. The gases leaving the stack included some ^{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 and reducing wall temperatures and reducing 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-gallon storage tanks. HLLW typically contained 300 Ci m^{-3}.

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

The NWCF was also the location of a decontamination facility used for cleanup of radiologically contaminated materials from INTEC and occasionally from other INEEL processes. The decontamination facility continues to function to support INEEL 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.

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^{-1} beta/gamma for brief periods. External exposure continues to accumulate from the decontamination facility.

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

FAST has two parts; a spent fuel storage area and the Fluorinel Dissolution Facility (FDF). The storage area consists of six stainless steel lined storage pools for storing spent nuclear fuel. The FDF

includes a hot cell with 6-foot-thick concrete walls where spent fuel was dissolved in an acid solution. The fluorinel process was used on zircalloy 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 1546 kg of uranium was reprocessed using the fluorinel dissolution process.

Internal exposure potential existed from airborne radioactivity that may have occurred from the various processes associated with the FAST.

External exposure occurred from unloading and loading irradiated fuel and fuel element examination under water and from work in the FDF hot cell environment which created radiation exposure up to 50 rad hr^{-1} for brief periods

2.3.7 INEEL Comprehensive Environmental Response Compensation and Liability Act Disposal Facility Complex, July 2003 to present

The INEEL 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 INEEL 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 (SSSTF). 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 (CAMU) 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 in the evaporation pond in accordance with the ICDF evaporation pond Waste Acceptance Criteria (WAC).

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

External exposure occurs when shipments of radioactive materials consisting of mixed fission products and activation products are placed in storage at the facility. Normal radiation levels are not permitted to be greater than 200 mrem hr^{-1} at any edge of the transporting vehicle. Higher radiation fields might be permitted, under special conditions, by proper management authority.

2.3.8 TMI-2 Independent Spent Fuel Storage Installation – CPP-1774 to present

The Independent Spent Fuel Storage Installation (ISFSI) is a new Nuclear Regulatory Commission (NRC)-licensed dry storage area for spent fuel and debris from the TMI accident. Fuel and debris were transferred to the INEEL Test Area North for examination, study, and storage after the accident. The fuel and debris were transferred to the ISFSI, which provides safe, environmentally secure, above-ground 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.

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

2.4 ARGONNE NATIONAL LABORATORIES-WEST, February 1951 to present

Argonne National Laboratory - West (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 site is in the southwest portion of INEEL, approximately 18 miles via Highway 20/26 east of Arco, 40 miles via Highway 26 northwest of Blackfoot, or 50 miles 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 No 3 (ZPR-III) facilities.

The present ANL-W site is in the southeast portion of INEEL, about 35 miles 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.

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

EBR-I, the first reactor built at INEEL, 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 within 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 constructed 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, $t_{1/2}$ = 15 hr. The saturation level at full-power operation was approximately 24 µCi gm^{-1}. The second most significant activity was about 2 µCi gm^{-1} ^{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.

External exposure was received by personnel from mixed fission products and activation products during activities associated with reactor operation and maintenance.

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

The Boiling Water Reactor Experiment No. 1 (BORAX-I) was an open-top, water-cooled, water-moderated, boiling-water reactor used to conduct a series of nondestructive experiments in the latter part of 1953 and early summer of 1954. The reactor was built in an excavated area of earth approximately one-half mile northwest of EBR-I and was housed in a 10-foot-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 feet by 350 feet south of the reactor area fence. Instrumentation at the control center showed an instantaneous radiation level higher than 500 mrem hr^{-1}, which decreased in about 30 seconds to 25 mrem hr^{-1} and within 5 minutes to less than 1 mrem hr^{-1}. A detailed discussion of this incident is available (Brodsky and Beard 1960).

Internal exposure may have occurred from airborne radioactivity during operation and other activities associated with a BWR, the core destruction, coolant, and clean-up activities.

External exposure occurred from direct radiation associated with the reactor operation maintenance activities from mixed fission products and activation products.

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

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. 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. 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. 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 boiling-water reactors.

Internal exposure was possible from MAP associated with work activities from the reactor coolant.

External exposure occurred during routine operations and with loading and unloading the reactor.

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

As a result of the BORAX-I and BORAX-II tests, a program began in March of 1955 to modify the BORAX-II reactor to produce electrical energy. 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 hours 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 at the NRTS, 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 MAP associated with work activities from the reactor coolant.

External exposure occurred from routine operation and loading and unloading of the reactor.

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. Like the previous BORAX reactors, BORAX-IV was a boiling-

water reactor 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 ºC 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. BORAX-IV released approximately 4,565 curies 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:

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 may have occurred during work with the defective fuel elements or during planned releases of short-lived fission activity and from airborne MAP/MFP associated with the reactor coolant.

External exposure occurred during operation and work with loading and unloading the reactor.

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

BORAX-V was a flexible boiling-water reactor, 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 may have occurred from coolant and airborne activity during routine BWR operation with fuel elements made from ceramics of uranium and thorium, and associated maintenance work.

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

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

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 x 42 ft x 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-curie polonium-beryllium neutron source. A hinged platform could be swung into place between the halves to stand on while loading or unloading the machine.

The storage vault room was approximately 29 feet long by 26 feet wide with walls and roof of reinforced concrete 9 inches thick. The fuel slugs were stored either on racks or in special "birdcage"-type containers that limited the storage density to 2 kilograms of ^{235}U per cubic foot.

Internal exposure potential existed from possible airborne radioactivity during maintenance operations from MFP/MAP.

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

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 kilowatt, was in a prefabricated Butler-type building built in 1958 near the EBR-I facility with its own heating and air compressor plant. 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 highly enriched uranium 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 may have occurred during routine operations that could create airborne radioactivity.

External exposure occurred from maintenance activities and loading and unloading fuel.

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 comprises a reactor building and a control building approximately 1 mile 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 atop the reactor during steady-state operations at 100 kW. Access to the subreactor room is controlled during steady-state operation. Prior to transient operations, the building was evacuated of all personnel. General neutron and gamma radiation levels at a distance of 10 feet from the reactor during operations at 100 kW were (Freund et al. 1960):

- Fast Neutron Negligible
- Thermal Neutrons 50-1,500 n cm^{-2} sec^{-1}

- Gamma 5-8 mrem hr^{-1}

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

External exposure occurred from routine operations.

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

EBR-II, at the ANL-W site, is a liquid sodium-cooled, unmoderated, heterogeneous fast breeder reactor rated at 62.5 MW thermal, with an intermediate closed loop of secondary sodium and a steam plant capable of producing electrical 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 that of a 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 gallons of liquid sodium. This tank was suspended in an airtight steel-shell containment building of 1-inch-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 (SPF) from its elemental, chemically unstable form, to a chemically stable composition suitable for landfill disposal.

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

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

2.4.11 Hot Fuel Examination Facility, 1964 to Present

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

Some 35,000 fuel elements were remotely reprocessed and refabricated into EBR-II subassemblies between 1964 and 1968. HFEF/South contains two large heavily shielded hot cells; one with an inert gas (argon) atmosphere, 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 post-irradiation 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 DOE's 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 (WCA) of the HFEF high bay.

Internal exposures might have occurred during cell entries when suspended radioactive contamination materials could cause airborne radioactivity from mixed fission products and activation products.

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

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

ZPPR is a split-table critical facility approximately 300 meters from EBR-II in the ANL-W area and about 3 miles 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-foot-diameter circular room with floor and walls of reinforced concrete. The roof is composed 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. 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.

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

2.4.13 Neutron Radiography Facility, October 1, 1977 to present

The Neutron Radiography (NRAD) Facility is a 250-kW, steady-state training research and isotope (built by 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 mixed fission products and activation products.

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

2.4.14 Fuel Assembly and Storage Building, 1970 to present

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 glove boxes and hoods. The facility no longer does fuel assembly, but other radiological work is ongoing.

Internal exposure may occur from airborne radioactivity associated with the described processes from the uranium fuel and spent fuel examinations.

External exposure occurred from movement of radiological samples and reactor fuel.

2.4.15 Other ANL-West Support Facilities

At ANL-W, an Analytical Laboratory (AL) provides the capability for performing chemical and physical measurements of both radioactive and non-radioactive samples. The facility includes six analytical hot cells and both 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 (RSWF) 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 (RLWTF) 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 (STP) north of EBR-II cover an area of about 2 acres. These ponds are not suspected to be radiologically contaminated.

Internal exposure is possible from airborne radioactivity associated with the various types of samples worked in these facilities.

External exposure occurred from working with the variety of radioactive material associated with these facilities (mixed fission products, activation products and transuranics).

2.5 RADIOACTIVE WASTE MANAGEMENT COMPLEX, MAY 1952 to present

The RWMC is 51 miles west of Idaho Falls. The first 13 acres were fenced off in May 1952 for shallow-land disposal of solid low-level radioactive waste, and burial of transuranic waste and hazardous substances such as organic and inorganic chemicals. The current RWMC mission includes interim storage of transuranic waste and shipment of stored transuranic 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, Rocky Flats began shipping defense waste with transuranic 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, transuranic waste was placed in retrievable storage on asphalt pads and covered with an earthen berm. This waste, in drums and boxes, is stored 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. INEEL has been repackaging (as needed) and shipping transuranic waste to the WIPP.

Early packaging configurations were thought 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^{-1} (photon energy = 250 keV).

In addition to administrative buildings, the RWMC has the following facilities and processes:

2.5.1 Subsurface Disposal Area

The 97-acre Subsurface Disposal Area (SDA) 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 transuranic and low-level waste was buried between 1954 and 1970. Solid waste from the Rocky Flats Plant in Colorado comprised a large fraction of the waste received at the RWMC. For example, in 1969 approximately 250,000 cubic feet of waste from the Rocky Flats Plant with a reported activity of more than 35,000 curies was buried at the RWMC. Rocky Flats waste was usually contaminated with plutonium isotopes and ^{241}Am (RAC 2002, pp. 26-27).

2.5.2 Intermediate Level Transuranic Storage Facility

The Intermediate Level Transuranic Storage Facility (ILTSF) has had 53 drums of ^{233}U stored in metal cargo containers in an open yard surrounded by concrete block shielding.

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 transuranic waste. This waste was received at INEEL after 1970 and was stored above ground.

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. Noncertified 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 transuranic waste into Transuranic Package Transporter (TRUPACT)-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 (AMWTP) under contract with DOE. The facility, which is under construction, will retrieve and process approximately 65,000 cubic meters of mixed transuranic 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.

Exposures below represent all of the subsections of 2.5.

Internal exposure may have resulted when workers were proximate to breached and or leaking waste containers containing mixed fission products, transuranics, activation products, etc. from the multiple materials in storage at the RWMC.

External exposure might have resulted from working adjacent to the various waste containers in storage or disposal in the form of intermediate-level, low-level, transuranic, and mixed waste, and from SWEPP X-ray (30-450 keV) and neutron (2-20 MeV) waste package examination and certification processes. Approximately 150 6M drums containing as much as 500 grams 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 mrem hr^{-1} with photon energies greater than 250 keV for short periods.

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, to present

The CFA is the main service and support center for INEEL programs and the other technical areas on the 890-square-mile Site. It is about 50 miles west of Idaho Falls, Idaho, just north of Highway 20 in

the south-central portion of the Site. Most 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 R&D work. What is now the INEEL 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 INEEL development that began in earnest in1950. The area continued to expand as a central service area for the NRTS. Eventually it acquired the name "Central," or more 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 (30-250 keV). The exposures listed below are for all of the subsections of section 2.6.

Internal exposure. CFA internal exposure potential was primarily from mixed fission products, activation products and transuranics associated with articles cleaned at the laundry facilities. Laundry facility workers were included in bioassay programs.

External exposure. CFA external exposure potential is greatest from calibration sources and X-ray equipment at the HPIL and the DOELAP Irradiation Facility with photon energies greater than 250 keV from calibration sources and 30-250 keV for X-ray photons. Neutron energies range from 2 to 20 MeV from the ^{252}Cf source and the AmBe source. Personnel working within radiological areas must wear dosimetry devices.

2.6.1 The Hot Laundry

The laundry, in the east portion of 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. Two laundry facilities next to one another (old and new facility) serviced INEEL. They are no longer operating.

2.6.2 The Health Physics Instrument Laboratory

The Health Physics Instrument Laboratory (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 health physics instrument calibrations. All of the sources were sealed. 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.

2.6.3 DOE Laboratory Accreditation Procedure Irradiation Facility

The DOELAP Irradiation Facility, CF-636, is an above-ground 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.

2.6.4 Radiological Environmental Science 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. 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 (CEL) 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-ID 2003).

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

The TRA is approximately 5 miles north of Central Facilities. 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 Measurements 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.

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 MTR (TRA 603) was the original reactor at the Test Reactor Area and the second reactor to be operated at INEEL. 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 once again fueled with ^{239}Pu. The last core was named "Phoenix" after the legendary bird that lived 500 years, burned itself to ashes, then rose to live again. The plutonium cores demonstrated that a plutonium-fueled, water-moderated reactor could be controlled satisfactorily. In August 1970, the MTR was again brought to power for a 24-hour run to irradiate 1,000 biological samples for iodine analysis.

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 mixed fission products and activation products from maintenance activities resulted in some potential for internal exposure. Some experiments in loops resulted in releases particularly of activation products.

External exposure occurred during the variety of operations associated with sampling a test reactor and the associated maintenance activities. The major contributors to external exposure were mixed fission and activation products that emitted beta and gamma radiation with energies typically above 250 keV.

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

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 older MTR and a more intense neutron flux. ETR fuel, coolant, and moderator materials were evaluated under environments similar to those of power reactors. Several experiment 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 documented immediately after shutdown.

Internal exposure from airborne radionuclides was minimal during normal reactor operation. Exposures may have occurred during shutdown as 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.

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^{-1} from fission and activation products in the reactor subpile room, loop cubicles, and nozzle trench.

2.7.3 Advanced Test Reactor, July 7, 1967, to present

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, ATR produces radioisotopes for use in medicine, industry, and for 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.

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 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.

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 watts. 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 mixed fission or activation products from airborne radioactivity during maintenance activities.

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.

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 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 International Business Machines (IBM) data cards. This refinement over the ARMF-I reactor meant that operators could process data quickly in electronic 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.

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.

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 calculational methods. The CFRMF contributed to the development of fast neutron reactors (Stacy 2000).

Internal exposure potential existed from airborne fission products that may have been released from the fuel.

External exposure. Minimal exposure resulted from working on the reactor top to move fuel or experimental components. Mixed fission or activation products would be the major contributors.

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. The ETRC enabled operators to predict the nuclear environment when completed experiments were removed or new ones added to calculate the experiment irradiation and determine core life, control rod withdrawal sequences, reactivity worth, and core safety requirements.

Proposed fuel and experiment loadings were mocked up in ETRC and 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 mixed fission products from damaged fuel or leaky experimental samples containing activation products is possible.

External exposure. Irradiated fuel or samples were transferred in or out of the ETRC under water to minimize exposure. The operating console was not on the reactor top, which also minimized external exposure. External exposure would have been from mixed fission products and activation products during loading or unloading fuel or test samples.

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

The ATRC performs functions for the ATR similar to those of the ARMF reactors in relation to 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 in or out of the ATRC. Some fuel or samples had been preirradiated in the ATR or other facilities.

External exposure occurred during canal work. The operators must on or leaning over the canal parapet to work with irradiated fuel or experiment samples being transferred in or out of the core under water. During reactor operation, access to the canal parapet is restricted. The ATRC canal is 21 feet deep.

2.7.10 TRA Hot Cell Facility 1954 to present

The TRA Hot Cells are southwest of the ETR reactor building. They consist of three separate cells with a common operating corridor. The operators are protected by thick concrete walls and special viewing windows, behind which they can handle, photograph, mill, measure, and weigh radioactive samples. The work in the cells has involved all types of samples including irradiated fuel, transuranics, 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.

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 transuranics, 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.

2.7.11 TRA Gamma Facility, 1955 to unknown

The TRA gamma facility was constructed south of the original TRA main security gatehouse. The facility consisted of a 16-foot-deep canal with cadmium buckets designed to hold spent MTR fuel elements. Experimental samples were inserted in sample tubes and lowered into extremely high gamma fields.

Sponsors have 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 is possible from a spent fuel element used for irradiations which is unlikely to be damaged.

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

2.7.12 Radiation Measurements Laboratory, 1952 to present

The Radiation Measurements Laboratory (RML), in the MTR west wing, was previously called the MTR Counting Room. The RML specializes in measuring quantity and quality of alpha, beta, gamma, and neutron radiation samples. A variety of counting equipment and spectrometers are available in the RML. Over the years, an endless variety of samples has been brought in for counting. Some of

the detector shielding was made out 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, transuranics, and more.

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.

2.7.13 Radiochemistry Laboratory

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 ways to produce and purify medical radioisotopes and the effects of radiation on hazardous waste.

Laboratories 109 through 112 were used primarily for chemistry analysis of reactor primary systems and loop experimental coolants. The predominant radioactivity was mixed fission and activation products. 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 transuranics, including ^{252}Cf.

Internal exposure potential existed from airborne radioactivity from the large variety of samples.

External exposures occur when irradiated samples are brought into the laboratories. One laboratory has a shielded box, similar to a small hot cell, used for handling highly radioactive samples.

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 gallons of wastewater per month were discharged to the seepage ponds and the ETR disposal well. Most of the activity pumped to the ponds is ^{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 (RAC 2002).

Internal exposure. The seepage ponds accumulated a significant quantity of mixed fission products. 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.

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.

2.7.15 High-Level Liquid Waste Tanks and Transfer Facility

On the south side of the MTR HP office, there are four 1,500-gallon 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 it is 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-gallon and two 9,000-gallon tanks.

Internal exposure potential exists from airborne radioactivity during entry to the holding tank pits for repairs from contaminants of mixed fission products and activation products.

External exposure due to sampling and transferring liquid waste is low because the tanks are in concrete-shielded underground vaults. Radiation sources are mixed fission or activation products when entry to the pits is necessary.

2.8 AUXILIARY REACTOR AREA, APRIL 1958 to late 1990s

The ARA was originally the Army Reactor Experimental Area (AREA), which was changed to the Army Reactor Area. It was established as a site 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 miles 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. 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.

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, boiling-water reactor designed and built by ANL applying its BORAX experience. The reactor, designed to generate only 1000 kilowatts, was originally named the Argonne Low Power Reactor. After ANL handed over the finished plant to the Army's operating contractor, Combustion Engineering (CE), 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 weeks, 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^{-1} around the ARA-II area. Initial recovery from the accident resulted in short-term exposure exceeding 500 rad $hr^{-1}$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 feet 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 contained in RAC (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 kilowatts 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 reactor 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.

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 feet away. It could be moved after a 36-hour shutdown. The reactor was designed for ease of operation and maintenance by technicians at remote installations, for reliable and continuous operation under extreme climatic conditions, and for the rigors of shipment and handling under adverse conditions. The Army phased out its reactor development program around 1965.

Exposures relate to all of the ARA facilities.

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 curies to the atmosphere). Airborne radioactivity consisting of mixed fission products and activation products would cause most internal exposure.

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 mixed fission products and activation products associated with the 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.

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

The Waste Reduction Operations Complex (WROC)/Power Burst Facility (PBF)/SPERT area is south-centrally located east of CFA on the INEEL site and 51 miles 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 INEEL radioactive, mixed, and low-level wastes. Many of the wastes treated originated at INEEL. Internal and external exposure will be noted at the end of section 2.9 for all the sub-sections.

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 feet in diameter and 14 feet 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.

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 pressurized-water reactor with coolant flow systems designed for operation with either light or heavy water. The pressure vessel was 24-1/2 feet high with a 10-foot 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.

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 gallons per minute, handle temperatures as high as 650°F, and pressures as high as 2,500 psi.

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 Driver 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 Loss-of-Coolant-Accident Program and the PBF. The SPERT reactors were decommissioned and replaced with other operations, as noted below.

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

The 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. Retrieving data and modification of the various test configurations resulted in exposure to high radiation fields and potential for release of fission products in the reactor containment.

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, into 1985 until 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 housed in the old SPERT-II facility. It is used to collect and store clean lead for the INEEL emergency lead inventory. The building was used in the past for storage of radioactive material. The area is not a radiological concern.

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 wastes for safe permanent disposition prior to 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 prior to disposal at the RWMC. In 1984, the incinerator began treating RCRA-defined mixed (radioactive and hazardous) waste. Wastes from INEEL and other DOE facilities were treated under provisions in the INEEL Site Treatment Plan. INEEL 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 (PCBs), corrosives, and flammables. Treatments are being developed for the types of wastes stored in the facility.

Internal exposures were possible based on releases from the various reactor operations at the SPERT reactors and PBF. Maintenance activities and other work with radioactive material (especially from PBF loop experiment) resulted in airborne mixed fission and activation products, making possible internal exposure with ^{137}Cs being the primary radionuclide.

External exposure resulted from experiment changes related to reactor experiment changes and maintenance activities. ^{137}Cs 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^{-1}. Other radiological work activities resulted in much lower exposure rates from the mixed fission products and activation products.

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.

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 what other reactor concepts had achieved 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 gallons of liquid consisting of diesel oil, xylene, methyl-chloroform, and a small amount of water were placed in an open vessel and ignited. Since the reactor did not operate, no other potential for internal dose occurred.

External exposures were possible from mixed fission products and activation products associated with core changes and associated reactor maintenance in radiological areas.

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 various tests (RAC 2002). These experiments were planned and conducted by the DOE HSL rather than by any contractor. Exposure information is summarized at the end of section 2.11 for the following sub-sections.

2.11.1 Fuel Element Burn Tests

Fuel Element Burn Tests A and B were conducted on test Grid III to support the General Electric 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^{-1} (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.

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 seconds during inversion conditions (Brodsky and Beard 1960). The test was conducted at 6:47 p.m. on March 20, 1957. FEBT-B was ranked above other release events at both onsite and offsite locations as an episodic event in RAC (2002).

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

Fission Products Field Release Tests (FPFRTs) 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 days and four were with fuel decayed for 42 to 65 days. 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 minutes, and maintained this temperature for approximately 10 minutes after melting began (Convair 1959). Instruments situated about a fan-shaped grid with seven concentric arcs and a maximum radius of about 5 miles obtained cloud diffusion, meteorological,

radiological, radiobiological, and deposition data. RAC (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 (RDTs) 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 RAC (2002).

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

The Experimental Cloud Exposure Study (EXCES) 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 out of 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 (RAC 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. The FEBT-B had a high ranking for INEEL releases, and the least amount of monitoring, according to RAC (2002). Personnel were involved in the Site's bioassay program.

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.

2.12 EXPERIMENTAL DAIRY FARM/EXPERIMENTAL FIELD STATION

The EFS was a 27-acre plot about 7 miles 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.

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. Specifically, these tests 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 the ICPP, ARA, NRF, and CFA areas. 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-minute period. Additional information regarding this test series is provided by Hawley (1964); Bunch (1966 and 1968), and Zimbrick and Voillequé (1969). Early in the test program, the AEC granted permission to six DOE-ID volunteers to be a 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.

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.

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

2.13 INEEL RESEARCH CENTER, 1984 to present

The INEEL 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 INEEL 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 INEEL 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 Pu, Sr, and Am sources for portable survey instrument response checks.

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

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.

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

The Army built the Army Reentry Vehicle Facility Site (ARVFS) 12 miles 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).

The ARVFS Facility 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 (RSAC)-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 Rhr^{-1}. The radiological fission product inventory in the NaK reported in LaRue and Dolenc (1986) was 23.3

grams and about 133 Ci. The Final Safety Analysis Report indicates that 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.

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

Table 2-3. ICPP 601/602 process cell information.

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-sir 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
M		
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	Health physics field office	
W	First cycle solvent raffinate	
X	Sample dilution & decontamination	
Y	Second & third cycle raffinate treatment	
Z	Product room	

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GLOSSARY

absorption
The process by which radiation imparts some or all of its energy to any material through which it passes.

activation
The process of inducing radioactivity by irradiation.

AmBe
A 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.

Albedo dosimeter
A device to detect and measure slow neutrons generated by higher energy neutrons incident on the body and that reflect back into the dosimeter.

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

anti-Cs
Anticontamination clothing, referring to special clothing worn by personnel for protection from radiological contamination.

Atomic Energy Commission
An agency established by the U.S. Government for oversight of nuclear weapons and power production; a predecessor to the U.S. Department of Energy.

background radiation
The radiation in an ambient environment which includes cosmic rays, radiation from natural sources, and man-made sources.

beta (ß) dose
A designation for external dose referring to the dose from less-energetic beta, X ray, and/or gamma radiation; typically a shallow dose or dose to the lens of the eye.

beta radiation
Radiation consisting of electrons or positrons emitted at high velocity from the nuclei of certain radioactive elements. Most direct fission products emit beta radiation.

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

boiling-water reactor
A nuclear reactor concept in which the coolant, water, is permitted to boil as it absorbs the heat of the nuclear reaction.

breeder reactor
A nuclear reactor concept in which the operation produces a net increase in fissionable reactor fuel.

calcine
The dry solid (grainy or granular) product of a chemical process of removing liquids from a solution; also, the process for creating the chemical reaction that removes liquids from a solution.

cladding
The outer layer of material encasing a reactor fuel element (e.g., aluminum or zirconium). Cladding promotes the transfer of heat from the fuel to the coolant and contains fission products and activation products that result from the fissioning of the fuel.

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

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

containment building
A safety feature of reactors typically engineered to be an airtight building, to prevent the release of radioactive gases or radiological contamination to the atmosphere or area outside the containment.

control rod
A device manipulated within a nuclear reactor constructed of material to absorb neutrons for the purpose of slowing down or increasing the nuclear reaction.

contamination, radioactive
Particulate matter on surfaces or in the air that is radioactive.

control room
The operating center of a nuclear reactor from which the reactor is operated and monitored.

coolant
In a reactor, a gas or fluid (water, liquid metal, etc.) contacting reactor fuel for the purpose of reducing or removing heat generated by the nuclear reaction.

core
That part of the reactor consisting of the fuel and some of the control elements for reactor operation.

criticality
A reaction capable of sustaining a nuclear chain reaction.

Curie
A special unit of radioactivity equal to 3.7×10^{10} disintegrations per second (dps).

D&D

Decontamination and decommissioning; dismantling or demolishing a facility after removal of radioactive and other potentially hazardous materials from the structure and premises.

decontaminate

A process removing radioactive particles from a person, place, or object.

depleted uranium

Uranium nuclide that has undergone a process to remove ^{235}U resulting primarily in ^{238}U.

dose equivalent (H)

The product of the absorbed dose (D), the quality factor (Q), and any other modifying factors. The dose unit is the rem.

dose

A specific amount of energy from ionizing radiation or a toxic substance absorbed per unit of mass.

dose of record

The dose recorded for individual worker files.

dosimeter

A device used to measure accumulated radiation exposure.

dosimetry

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

Emergency Core Cooling System

An emergency backup system designed to inject coolant into the core of a reactor to prevent the overheating of the fuel and subsequent fuel damage.

enriched uranium

Uranium enhanced from its natural state to contain a higher abundance of the isotope ^{235}U.

excursion

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

exposure

Technically, a measure of X-ray or gamma radiation capability to ionize air (units of Roentgen).

extremity

The arm from and including the elbow through the fingertips and the leg extending from and including the knee through the toes.

film

Generally means a "film packet" that contains one or more pieces of film in a light-tight wrapping. The developed film has an image caused by radiation that can be measured using an optical densitometer.

fission

A nuclear transformation characterized by the splitting of a nucleus into at least two other nuclei and the release of a relatively large amount of energy.

fission product

Elements or compounds resulting from fission.

flux

The intensity of particles (e.g., the number of neutrons passing through a unit area in unit time.)

fuel assembly

An arrangement of nuclear fuel and its cladding material into a particular form and shape for use in a nuclear reactor. Fuel may be assembled in plates, rods of various diameters, or other shapes.

fuel reprocessing

A chemical process, usually involving several steps, that recovers ^{235}U and other fissionable products from spent fuel.

gamma rays

Short wave length electromagnetic radiation (photons) originating in atomic nuclei and accompanying many nuclear reactions (e.g., fission, radioactive decay, and neutron capture) in an energy range of 10 keV to 9 MeV.

Gray (Gy)

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

half-life

The time it takes for one-half of any given number of unstable atoms to decay (disintegrate).

HEU

Highly enriched uranium.

hot cell

A 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

An operational (or test) run of a chemical process and equipment using radioactive materials.

interim storage

A concept in the management of nuclear waste in which the waste is moved to an intermediary location between its point of origin and its ultimate storage location.

ionization chamber

A device used to measure exposure or radiation dose.

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

irradiate
To expose a substance to radiation.

isotope
Nuclides having the same number of protons in their nuclei (same atomic number), but having a differing number of neutrons (different mass number).

megawatt
A measure of electrical power equal to 1 million watts.

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

millirem
A unit of radiation equal to one-thousandth of a rem (see rem).

microcurie
A measure of radioactivity equal to one-millionth of a curie.

mixed waste
Waste that is both chemically hazardous and radioactive.

moderator
A material used in a nuclear reactor to reduce the natural speed of neutrons ejected from fissioning atoms (water, graphite etc.).

natural uranium
Uranium occurring in nature that has not been through an enrichment process.

neutron
A basic particle in a nuclear reaction, electrically neutral, with nearly the same mass as a hydrogen atom.

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

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

nuclear energy
Energy released from a nuclear fission or fusion reaction.

nuclear power plant
An electrical generating facility using nuclear fuel.

nuclear waste
A general term used for the byproduct nonuseable material resulting from nuclear reactions, including high-level, intermediate, low-level, mixed and transuranic waste.

nucleus
That part of an atom consisting of the total positive electrical charge and most of the mass.

pocket dosimeters
A type of ionization chamber used by personnel to measure radiation dose. Other names are pencil dosimeter, pocket pencil, pocket ionization chamber (PIC).

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

photon
A quantum of electromagnetic energy often referred to as X-rays or gamma rays, but also including light and radiant heat.

Pressurized-water reactor
A concept in which water used to cool the reactor core is pressurized to prevent boiling. Heat is typically transferred from a primary system to a secondary system.

primary loop
A 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.

proton
An elementary atomic particle with a positive electrical charge equal numerically to the charge of the electron and a mass slightly greater than 1 mass unit.

Quality factor, Q
A modifying factor used to derive dose equivalent from absorbed dose.

rad
The unit of absorbed dose.

radiation
Energy transferred through air or some other media in the form of particles or waves (see ionizing radiation).

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

radioactive waste
Byproducts of nuclear processes that are radioactive and have no useful recyclable purpose (see *nuclear waste*).

radionuclide
A radioactive species of an atom characterized by the constitution of its nucleus specified by the number of protons, neutrons, atomic number, and mass number.

RaLa
Radioactive lanthanum, one of the fission products of a nuclear reaction; a lanthanum recovery process at the INTEC for development weapons.

reactor vessel
A steel container enclosing the fuel elements, control elements, coolant piping, and other structures that support the core of a nuclear reactor.

reflector
Part of the structure of some nuclear reactors designed to reflect neutrons back into the core of the reactor.

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

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

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

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

secondary loop
The system in a reactor receiving transfer heat from a primary system. The secondary system is not exposed to the reactor core and is free of radioactivity under normal circumstances.

semiscale
The informal name of a scale model of a nuclear reactor operated at INEEL. 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
The 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.

shielding
Any material or obstruction that absorbs (or attenuates) radiation to protect personnel or materials from radiation.

Sievert (Sv)
The SI unit for dose equivalent (1 Sv = 100 rem).

skin dose
Absorbed dose at a tissue depth of 7 mg cm^{-2} ~ 0.07 mm in tissue.

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

spent fuel storage basin
A pool or pit made of reinforced concrete containing water and used to store spent nuclear fuel. The water acts as shielding and as a coolant.

thermoluminescence
Property of a material resulting in light emission caused from excitation from heat.

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

transuranic (TRU) waste
Contaminated waste materials with nuclides having an atomic number greater than 92, a half-life over 20 years and concentration greater than or equal to 100 nCi $gram^{-1}$.

TRIGA
An acronym for a type of training research reactor built by General Atomics.

tritium
A radioactive isotope of hydrogen having one proton and two neutrons. Tritium gas is produced in nuclear reactors and used to boost the explosive power of most modern nuclear weapons. It is also a constituent of irradiated water associated with reactor operations.

uranium-235
A fissionable uranium isotope enriched for use as a fuel in nuclear reactors.

uranium-238
The most common uranium isotope, typically nonfissioning; can be irradiated in a reactor and transformed to plutonium for use as fuel.

uranium oxide
A 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.

water-moderated reactor
A reactor concept designed so that water slows down the speed of neutrons from fissioning atoms.

waste storage tank
A holding tank for liquid or gaseous wastes which might or might not be radioactive.

whole-body dose

Commonly defined as the absorbed dose at a tissue depth of 1.0 cm (1000 mg/cm^2); used to refer to the recorded dose.

X-ray

Ionizing electromagnetic radiation of external nuclear origin with energies les than 250 keV or a radiograph.

zirconium

A metallic element highly resistant to corrosion and often used to make cladding for nuclear fuel. It is sometimes alloyed in small amounts in the fuel itself.

zero power

Also called low power; operating a reactor to maintain a chain reaction at an extremely low power level producing very little heat. Zero power reactors are used as sensitive laboratory tools to pretest experimental loadings of test reactors and for other analytical purposes.