Approvals	Supersedes:
Subject Expert: Henry Peterson Document Owner Approval: Signature on File Date: 09/27/2004 Norman D. Rohrig, TBD Team Leader Approval: Signature on File Date: 09/27/2004 Judson L. Kenoyer, Task 3 Manager Concurrence: Signature on File Date: 09/28/2004 Richard E. Toohey, Project Director Approval: Signature on File Date: 09/30/2004 James W. Neton, Associate Director for Science	None

TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

RECORD OF ISSUE/REVISIONS

ISSUE AUTHORIZATION DATE	EFFECTIVE DATE	REV. NO.	DESCRIPTION
Draft	07/02/2004	00-A	New Technical Basis Document for the Argonne National Laboratory – Occupational Environmental Dose. Initiated by Norman D. Rohrig.
Draft	08/25/2004	00-B	Incorporates NIOSH and internal review comments. Initiated by Norman D. Rohrig.
09/30/2004	09/30/2004	00	First approved issue. Initiated by Norman D. Rohrig.

ACRONYMS AND ABBREVIATIONS

AEC	U.S. Atomic Energy Commission
AFSR	Argonne Fast Source Reactor
ANL-W	Argonne National Laboratory-West
ARA	Auxiliary Reactor Area
ATR	Advanced Test Reactor
BORAX	Boiling Water Reactor Experiment
Bq	Becquerel (1 disintegration per second)
CDC	Centers for Disease Control and Prevention
CERT	Controlled Environmental Release Test
CFA	Central Facilities Area
CFR	Code of Federal Regulations
Ci	curie
CPP	(Idaho) Chemical Processing Plant
DOE	U.S. Department of Energy
EBR-I	Experimental Breeder Reactor No. 1
EBR-II	Experimental Breeder Reactor No. 2
EFS	Experimental Field Station
EMDR	Environmental Monitoring Data Report
EMR	Environmental Monitoring Report
ETR	Engineering Test Reactor
F	Fast (solubility rate)
fCi	femtocurie
FECF	Fuel Element Cutting Facility
FP	Fission Product
FPFRT	Fission Product Field Release Test
GE	General Electric (Company)
GE-ANP	General Electric-Advanced Nuclear Propulsion (Program)
H&S	Health & Safety (management organization)
HTRE	Heat Transfer Reactor Experiment
ICPP	Idaho Chemical Processing Plant
IDO, ID	Idaho Operations Office (of AEC)
IET	Initial Engine Test
INEEL	Idaho National Engineering and Environmental Laboratory
INEL	Idaho National Engineering Laboratory
INELHDE	Idaho National Engineering Laboratory Historical Dose Evaluation
INTEC	Idaho Nuclear Technology and Engineering Center
kW	kilowatt
M	Moderate (solubility rate)
mR	milliRoentgen

mrem	millirem
MTR	Materials Testing Reactor
NCRP	National Council on Radiation Protection and Measurements
NOAA	National Oceanic and Atmospheric Administration
NRTS	National Reactor Testing Station
RAC	Radiological Assessment Corporation
RM	radioactive material
RSAC	Radiological Safety Analysis Computer (program)
RWMC	Radioactive Waste Management Complex
S	Slow (solubility rate)
SL-1	Stationary Low-Power Reactor No. 1
TAN	Test Area North
TBD	technical basis document
TLD	thermoluminescent dosimeter
TRA	Test Reactor Area
TREAT	Transient Reactor Test
U.S.C.	United States Code
ZPR	Zero Power Reactor
ZPPR	Zero Power Plutonium Reactor

4.1 INTRODUCTION

Technical Basis Documents (TBDs) 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 the National Institute for Occupational Safety and Health 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 Energy Employee Occupational Illness Compensation Program Act of 2000 (42 U.S.C. § 7384l (5) and (12)).

This TBD addresses radioactive material (RM) releases from areas or facilities at the Idaho National Engineering and Environmental Laboratory (INEEL), formerly the National Reactor Testing Station (NRTS) and later the Idaho National Engineering Laboratory (INEL), that could affect employees at the Argonne National Laboratory-West (ANL-W) facility. The analysis for this TBD divided releases into two components: (1) normal “chronic” operational releases, and (2) episodic releases that generally are of short duration. These releases potentially represent unrecorded or missed doses, as either direct gamma radiation or beta-gamma radiation from immersion in the radioactive gaseous cloud, for individuals who did not have personal dosimetry to record the dose, or as internal doses from RM inhalation.

In addition, this TBD addresses direct gamma doses resulting from facility operations. In general, these doses, if not controlled by management, increase with time and create a facility background dose. At INEEL, these *facility background doses* were recorded by film badges infrequently and inconsistently before 1970 and by thermoluminescent dosimeters (TLDs) on a routine basis since 1972. These facility background doses, or facility *fence-line* doses, as they are sometimes called, are a nebulous indication of a dose that workers could receive if they inhabited outside areas within the facility. This TBD presents ANL-W facility fence-line doses (minus background) for three locations: 1) the EBR-I location for 1952 to 1972, 2) the Transient Reactor Test (TREAT) facility and the Experimental Breeder Reactor (EBR)-II both for 1972 to present.

As outlined and discussed in Part 2 of the ANL-W facility profile (ORAU 2004), the U.S. Atomic Energy Commission (AEC) selected the INEEL Site as an isolated location for testing various reactor concepts. The INEEL is isolated from the public in two important aspects: (1) it is remote from population centers, and (2) it is isolated hydrologically because no surface streams originate on the Site and flow to an offsite location and no streams cross the Site. Although the INEEL sits above the large Snake River Aquifer that eventually surfaces and enters the Snake River in the Hagerman Valley area, the annual flow rate of the water in the aquifer is 5 to 15 feet per day.

ANL-W is a unique facility at the INEEL. Although inside the INEEL boundary, ANL-W is under the jurisdiction of the U.S. Department of Energy (DOE) Chicago Operations Office. Although the facility operates in accordance with 10 CFR 835, its operations have been and are more in line with a university atmosphere engaged with pure nuclear energy research; these operations support those of the University of Chicago or ANL-E, near Chicago, Illinois. During its first 14 years of operation (i.e., from 1951 through 1965), ANL-W was in the southwest corner of INEEL at the location of EBR-I. At this location ANL-W conducted EBR-I, Zero Power Reactor No. 3 (ZPR-III), Argonne Fast Source Reactor (AFSR), and all Boiling Water Reactor Experiments (BORAX) tests. In 1958, construction began on the TREAT Facility and on the EBR-II at the location presently designated as ANL-W at the southeastern corner of INEEL. Since the mid-1970s, essentially all ANL-W operations have been conducted at the present ANL-W location, as shown on Figure 4-1.

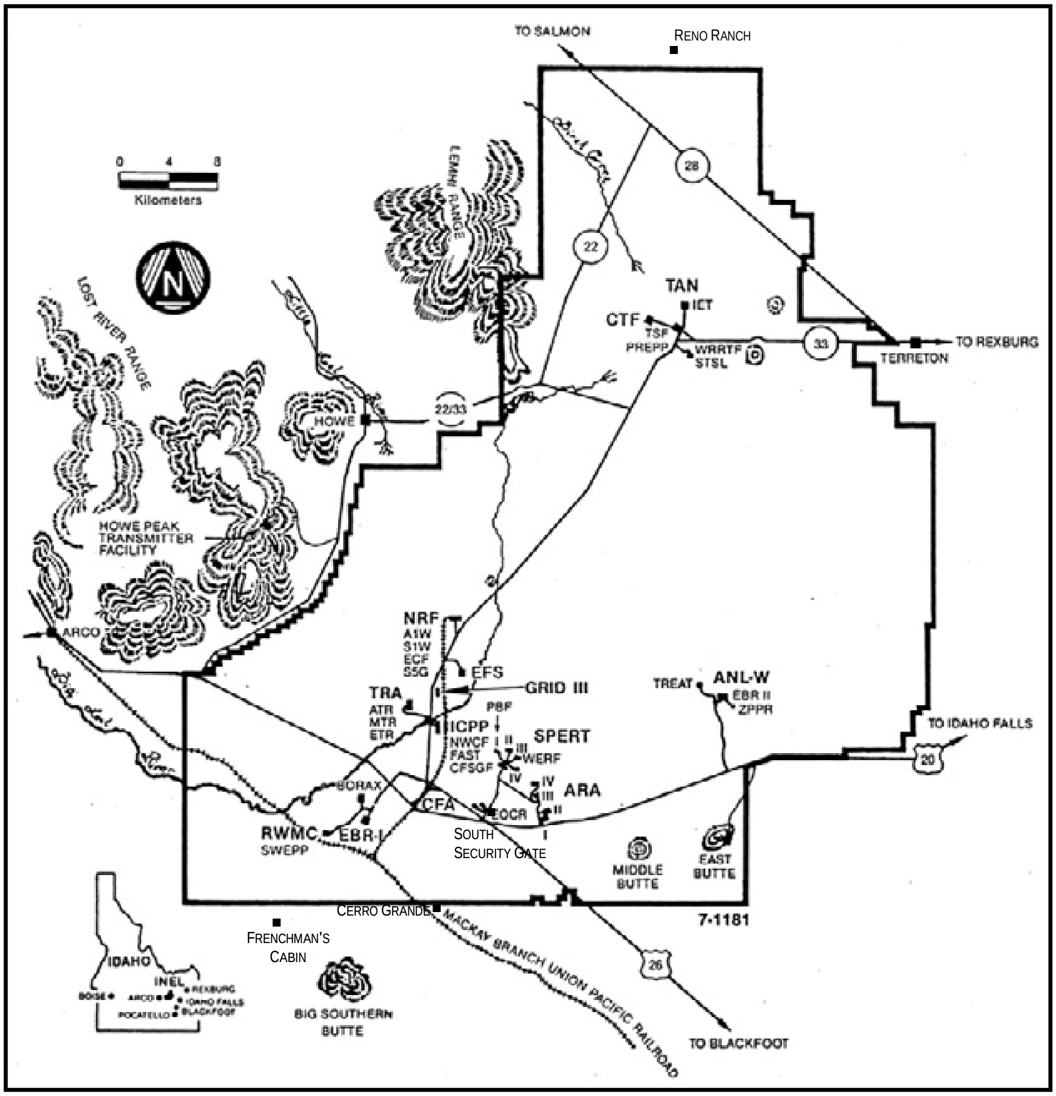

Figure 4-1. INEEL site map.

During the 50-year history of the INEEL Site, DOE and its predecessor agencies designed, built, and operated about 50 different reactor concepts at INEEL. All of these reactors have been prototype, low-power critical, or test reactors. INEEL operated no weapons production or commercial power reactors. Most, if not all, of these reactors have used highly enriched (93% or higher) uranium as fuel. Only a few have produced significant airborne effluent: (1) the Heat Transfer Reactor Experiment (HTRE) reactors, operated under the General Electric-Aircraft Nuclear Propulsion (GE-ANP) Program at the north end of the INEEL Site at Test Area North (TAN), (2) test reactors [Materials Testing Reactor (MTR), Engineering Test Reactor (ETR), and Advanced Test Reactor (ATR)], all at the Test Reactor Area (TRA) near the middle southern end of the Site, and (3) EBR-II at ANL-W at the southeastern corner of the Site, which has produced minor amounts of airborne effluent.

Another historically important airborne effluent producer is the Idaho Nuclear Technology and Engineering Center (INTEC), formerly known as the Idaho Chemical Processing Plant (ICPP). This facility, constructed in the early 1950s, began processing nuclear fuel in February 1953 and continued until 1992. Throughout its history, the "*Chem" Plant*, as it is commonly known, has reprocessed fuel from test reactors at INEEL, zirconium-clad fuel reclaimed from various reactors, stainless-clad fuel from EBR-II, and many AEC test reactors from around the world. Apart from the GE-ANP Program, which tested nuclear powered aircraft engine concepts with only one barrier (fuel cladding) between the fission products and the environment, TRA and ICPP airborne releases have been the most radiologically significant releases at INEEL (RAC 2002). Through the years that INEEL has published environmental monitoring reports, ICPP airborne effluents have been attributed to creating the majority of the INEEL "boundary dose." Considering this fact, dose reconstructors should suspect ICPP airborne effluent as responsible for the maximum INEEL worker doses. Calculations performed for the INEEL TBD show that although ICPP airborne effluent is the most radiologically significant release at INEEL, the impact to all facility workers at INEEL is significantly below the allowable and acceptable limit. Figure 4-1 shows INEEL facility locations, including the EBR-I and ANL-W facilities.

From the beginning of operations at the INEEL Site, DOE and its predecessor agencies selected facility locations to limit the potential for operational releases at one facility to affect another facility. Because the Site encompasses 890 square miles, there was ample room to place facilities with this principle in mind. Because the Site has an average elevation of 5,000 feet and its general meteorological characterization indicates a nocturnal inversion from the north-northeast and a daytime lapse condition with winds from the southwest, transitional weather regimes are less frequent than at lower elevations. The 50-year history of the Site has demonstrated that the large expanse of INEEL and this meteorological characteristic have been effective in maintaining the operational isolation of each facility.

Beginning with the GE-ANP program, which began in the early 1950s, the INEEL Site has had the capability of plume tracking by aircraft. The local National Oceanic and Atmospheric Administration (NOAA) field office, which was dedicated to INEEL needs and requirements, provided plume projection capabilities for various programs with a rather extensive network of meteorological monitoring stations (Yanskey, Markee, and Richter 1966). The plumes from all intentional planned releases from the GE-ANP tests, the Controlled Environmental Release Test (CERT), the Fission Product Field Release Test (FPFRT), the Fuel Element Burn Tests A and B, etc., were directed over an instrumented monitoring grid (GRID III) that was remote from other facilities, such that releases did not affect other onsite facilities.

INEEL reviewed and analyzed all airborne releases that have occurred since the beginning of Site operations as a result of a request from the DOE Idaho Operations Office (IDO) to evaluate the radiological impact to individuals at the INEEL boundary from airborne releases that had occurred since the beginning of Site operations. With the help of NOAA, which had hourly meteorological data from 1956 to that time, INEEL completed analyses for all airborne releases that occurred at the Site. Radiological consequences for an adult, a child, and an infant were calculated with Version 4 of the Radiological Safety Analysis Computer program RSAC-4 (Wenzel 1990). The results of the study were published in the *Idaho National Engineering Laboratory Historical Dose Evaluation* (DOE 1991); this TBD refers to that report as the INELHDE. All releases considered for that report are the bases for the releases considered in this TBD. In addition, all releases documented in the INELHDE, operational and episodic, have been independently reviewed and found, with minor modifications, to be substantially appropriate. The review, conducted by Radiological Assessment Corporation (RAC 2002) at the request of the Centers for Disease Control and Prevention (CDC) and the State of Idaho, evaluated the methodology by which the RSAC-4 computer program performs dose calculations against the methodology favored by the National Council on Radiological Protection and Measurements (NCRP). It stated: "*As a final point, Tables 7, 8, 9a, 9b, 10a, and 10b, and Figures 18 and 19 confirm that the NCRP method was suitable for these ranking purposes when the results are* *compared with those using the RSAC code. In all cases, the RSAC code confirmed the results obtained using the NCRP methodology"* (RAC 2002, p. 57).

Version 6 of the RSAC code (Wenzel and Schrader 2001) is used extensively in this document to provide onsite concentrations due to episodic releases as well as other evaluations. For more information on the RSAC code, see Peterson (2004).

EBR-I was the first reactor to operate at the Site. It and BORAX-I through –V were in the southwestern corner of INEEL, operated under the AEC Chicago Operations Office by the University of Chicago as ANL-W. These low-power reactors produced essentially no radioactive airborne effluent. As the EBR-I and BORAX programs ended, ANL-W relocated the locations of EBR-II, the TREAT Facility, the Zero Power Plutonium Reactor (ZPPR), etc., to the eastern section of INEEL. The EBR-I location is now a historic landmark and is open to the public during the summer months.

All inhaled quantities and concentrations referred to in this TBD apply to individuals stationed at the ANL-W facility, at the South-West location through 1965, and at the South-East location beginning in 1966.

4.2 INTERNAL INTAKES FROM ONSITE AIRBORNE RADIONUCLIDE CONCENTRATION

This section addresses onsite concentrations of radionuclides and onsite internal personnel intakes from normal operational releases and from shorter term releases such as those from criticality incidents at ICPP and tests performed by the GE-ANP Program at TAN. As stated above, operational releases from ICPP and TRA have been the predominant and radiologically significant releases at INEEL during the history of the Site. For more discussion of these releases and their relationship to other, less significant releases, see Peterson (2004), DOE (1991), or RAC 2002.

For worker dose reconstruction, the analyst should use default values for the calculation under consideration. When solubility is of concern, an insoluble oxide form for solids is recommended for analysis, with "S" and "M" type materials being the predominant form. Without more definitive information on the type of material, the dose reconstructor should use the material that maximizes the dose for a particular situation. When iodines are of concern, the dose reconstructor should consider them to be "F" type materials.

4.2.1 Operational Releases

Estimation of onsite concentrations of radionuclides and resulting potential intakes from operational releases at INEEL facility locations employs the same methodology used to determine offsite concentrations for Site annual environmental monitoring reports. The release for each year of operation is exactly the same as that documented in INELHDE (DOE 1991) with one exception: an analysis to reduce the number of radionuclides and yet retain those that contributed about 95% of the inhalation dose (see Peterson 2004) reduced the number of radionuclides from 56 to 9 for the operational releases.

Meteorological dispersion factors applicable to each INEEL facility were picked from the annual average mesoscale dispersion isopleths of ground-level air concentrations as published in environmental monitoring reports, as described in INELHDE (DOE 1991). As described in Appendix B of that document, dispersion isopleths are available for years beginning in 1973, with the exception of 1978, when INEEL upgraded the telemetry system. For years prior to 1973, the TBD analysis used a 9-year average of mesoscale dispersion isopleths of ground-level air concentrations (Bowman

1984), shown in Figure 4-2. For 1993 to 2002, the analysis used annual average mesoscale isopleths from the annual environmental reports (ESRF 1994, 1995, 1996, 1997, 1998, 2000; Stoller 2002a,b,c) to calculate the facility annual concentration.

Of the many facilities on INEEL, this analysis considered only the ANL-W facility. Yearly isopleth values for each ANL-W location (EBR-I for 1952 to 1965 and EBR-II for 1966 to present) have been extracted from the annual environmental monitoring reports and converted from the normalized annual concentration[1] (hr^2/m^3) to concentrations (Bq/m^3)

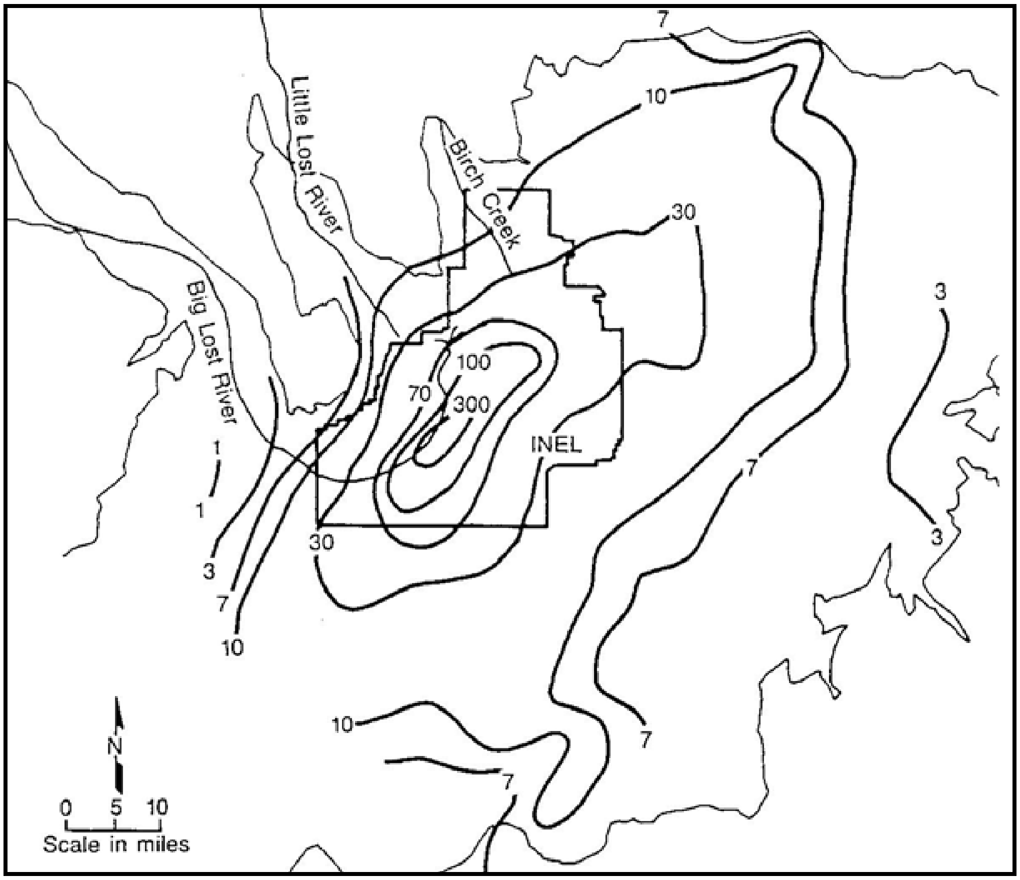

Figure 4-2. Nine-year (1974 - 1983) average mesoscale dispersion isopleths of air concentration at ground level ($hr^2/m^3 \times 10^{-9}$), normalized to unit release rate (Bowman 1984).

and multiplied by 2.4×10^3 m^3/yr (the amount of air breathed occupationally per year) to produce activity inhaled per year (Bq) for an occupational individual. Table 4-1 lists these values.

The annual inhaled quantities (Bq/yr) listed in Table 4-1 are based on known and published INEEL annual airborne emissions. The following discussion provides information that is located in NRTS/INEL/INEEL documentation on facility environmental sampling/monitoring and data that can be compared with the calculated inhaled quantities of Table 4-1. “Environmental” air sampling at the facility areas has been performed at least since the mid-1950s where airborne effluents were known or suspected to exist. The early IDO Health and Safety (H&S) Division Annual Reports document many studies for defining radionuclide concentrations in the vicinity of different facilities. These

[1] As used at the INEEL, this quantity is the sum of 8,766 calculations of the hourly average χ/Q.

Table 4-1. Intake (Bq yr^{-1}) by year for ANL-W, 1952-2002.

Nuclide	Ce-144	I-131	Pm-147	Pu-238	Pu-239, 240	Ru-106	Sr-89	Sr-90	Y-91
1952	2.4E+1	1.4E-1	0.0E+0	0.0E+0	0.0E+0	1.7E+0	0.0E+0	5.4E-1	2.2E+0
1953	2.4E+1	3.1E-1	5.7E+0	3.8E-3	5.6E-4	1.7E+0	1.7E+0	2.9E+0	3.7E+0
1954	6.0E+1	2.3E-1	1.4E+1	9.5E-3	1.4E-3	4.4E+0	4.3E+0	5.1E+0	9.2E+0
1955	8.3E+1	3.4E-1	2.0E+1	9.5E-3	1.4E-3	6.1E+0	6.0E+0	7.3E+0	1.3E+1
1956	9.6E+1	4.0E+0	2.3E+1	1.5E-2	2.3E-3	7.0E+0	7.2E+0	7.6E+0	1.5E+1
1957	1.9E+1	4.8E+2	4.8E+1	2.7E-2	4.0E-3	2.0E+0	6.6E+0	1.2E+1	7.3E+0
1958	2.6E+1	3.6E+2	6.9E+1	3.9E-2	5.7E-3	2.8E+0	5.1E+0	1.8E+1	5.6E+0
1959	2.1E+1	7.8E+1	5.7E+1	3.1E-2	4.6E-3	2.3E+0	2.6E+0	1.5E+1	2.9E+0
1960	2.8E-1	1.1E+1	6.0E-1	1.1E-2	1.6E-3	2.8E-2	3.2E-1	9.0E-1	3.5E-1
1961	2.0E-1	1.5E+1	1.4E-2	2.0E-3	2.9E-4	1.2E-2	8.5E-1	1.1E+0	9.2E-1
1962	8.9E-1	1.4E+1	2.0E+0	1.3E-3	1.9E-4	8.9E-2	8.6E-1	1.6E+0	9.3E-1
1963	1.4E+1	9.2E+0	4.0E+1	1.2E-2	1.8E-3	1.6E+0	5.4E-1	1.2E+1	6.0E-1
1964	7.8E+0	4.8E-1	0.0E+0	4.7E-4	6.9E-5	1.2E+2	1.3E-1	3.0E+0	8.5E+0
1965	2.0E+1	3.2E+0	0.0E+0	2.0E-2	3.0E-3	8.5E+0	0.0E+0	1.2E+1	7.5E+0
1966	2.8E+0	4.3E-1	0.0E+0	1.1E-3	1.6E-4	1.2E+1	0.0E+0	7.8E-1	1.2E+0
1967	7.0E-2	1.8E-1	0.0E+0	1.2E-4	1.8E-5	1.7E+0	0.0E+0	2.1E-1	6.6E-1
1968	5.0E+0	3.4E-1	0.0E+0	2.3E-3	3.4E-4	7.4E-1	0.0E+0	1.2E+0	6.4E-1
1969	2.8E-1	4.8E-1	0.0E+0	5.0E-4	7.4E-5	3.7E-1	0.0E+0	3.6E-1	5.3E-1
1970	6.6E-1	2.5E-5	0.0E+0	7.0E-4	1.1E-4	3.0E-1	0.0E+0	2.7E-1	5.5E-1
1971	2.5E+0	7.0E-1	0.0E+0	2.1E-3	3.1E-4	3.2E+0	0.0E+0	1.1E+0	4.7E-1
1972	2.7E-1	2.9E-1	0.0E+0	6.5E-4	9.7E-5	4.9E-1	0.0E+0	2.8E-1	1.7E-1
1973	2.4E-2	1.4E-5	0.0E+0	2.4E-4	3.5E-5	1.1E-1	0.0E+0	6.2E-2	2.2E-2
1974	7.9E-3	1.3E-3	0.0E+0	1.1E-4	9.7E-6	4.4E-2	0.0E+0	3.7E-2	1.2E-1
1975	8.9E-3	4.5E-3	0.0E+0	1.3E-4	2.5E-5	6.4E-2	0.0E+0	1.9E-2	2.5E-1
1976	5.2E-5	3.2E-5	0.0E+0	8.1E-6	3.6E-6	8.1E-4	0.0E+0	3.9E-4	3.0E-2
1977	2.0E-4	1.4E-4	0.0E+0	8.0E-5	3.4E-5	1.1E-2	0.0E+0	5.0E-3	4.3E-1
1978	3.8E-4	2.0E-3	0.0E+0	7.4E-5	7.9E-6	5.7E-3	0.0E+0	1.9E-3	3.5E-1
1979	1.8E-4	9.7E-5	0.0E+0	4.8E-5	5.2E-6	1.3E-3	0.0E+0	8.9E-3	5.1E-2
1980	2.9E-4	1.5E-3	0.0E+0	3.1E-5	4.0E-6	6.3E-4	0.0E+0	4.3E-4	3.1E-1
1981	2.9E-4	3.8E-3	0.0E+0	6.1E-6	1.1E-6	6.1E-3	0.0E+0	3.3E-4	2.0E-1
1982	1.5E-4	4.7E-5	0.0E+0	1.5E-5	1.6E-6	4.4E-4	0.0E+0	2.8E-4	7.2E-2
1983	2.9E-4	1.5E-3	0.0E+0	1.2E-4	1.6E-5	2.3E-3	0.0E+0	1.1E-4	3.6E-2
1984	2.9E-4	9.7E-5	0.0E+0	1.9E-5	7.4E-6	3.2E-4	0.0E+0	1.3E-4	1.4E-2
1985	1.2E-3	9.0E-3	0.0E+0	2.3E-5	4.5E-6	1.0E-2	0.0E+0	6.6E-4	9.9E-1
1986	2.9E-4	8.9E-5	0.0E+0	1.3E-6	9.7E-8	2.3E-3	0.0E+0	1.6E-5	4.3E-2
1987	2.9E-4	4.3E-5	0.0E+0	1.4E-6	2.1E-7	3.0E-5	0.0E+0	2.3E-5	7.6E-1
1988	2.9E-4	1.4E-5	0.0E+0	1.1E-6	1.7E-7	1.5E-2	0.0E+0	2.7E-5	5.2E-1
1989	2.9E-4	9.7E-6	0.0E+0	5.7E-9	8.1E-10	1.6E-4	0.0E+0	7.4E-6	6.7E-2
1990	1.2E-4	3.8E-5	0.0E+0	9.4E-10	9.4E-10	4.2E-5	0.0E+0	2.0E-7	3.1E-2
1991	1.2E-4	1.6E-5	0.0E+0	1.0E-10	1.0E-10	5.2E-5	0.0E+0	9.7E-5	1.8E-2
1992	1.2E-4	1.8E-5	0.0E+0	1.7E-7	1.7E-7	1.4E-5	0.0E+0	8.3E-6	3.1E-2
1993	0.0E+0	3.8E-6	0.0E+0	0.0E+0	8.7E-11	3.5E-5	0.0E+0	3.8E-5	0.0E+0
1994	0.0E+0	3.1E-5	0.0E+0	0.0E+0	4.6E-8	0.0E+0	0.0E+0	8.0E-5	0.0E+0
1995	0.0E+0	2.1E-5	0.0E+0	3.3E-8	5.5E-9	0.0E+0	0.0E+0	2.3E-6	0.0E+0
1996	0.0E+0	2.8E-5	0.0E+0	2.2E-7	4.5E-9	0.0E+0	0.0E+0	1.1E-6	0.0E+0
1997	0.0E+0	0.0E+0	0.0E+0	1.8E-7	5.5E-8	0.0E+0	0.0E+0	2.4E-5	0.0E+0
1998	0.0E+0	2.3E-5	0.0E+0	1.7E-7	1.8E-8	0.0E+0	0.0E+0	1.1E-5	0.0E+0
1999	0.0E+0	3.1E-5	0.0E+0	7.5E-8	7.1E-9	0.0E+0	0.0E+0	4.4E-6	0.0E+0
2000	0.0E+0	1.9E-3	0.0E+0	3.6E-5	3.6E-7	0.0E+0	0.0E+0	3.5E-3	0.0E+0
2001	0.0E+0	1.0E-3	0.0E+0	2.4E-7	2.9E-5	0.0E+0	0.0E+0	1.2E-4	0.0E+0
2002	0.0E+0	1.6E-4	0.0E+0	6.6E-6	1.8E-5	0.0E+0	0.0E+0	3.8E-3	0.0E+0

studies were specific for a given test, operation, or incident, however, and did not occur in a set location or for a standard duration. The 1963 Annual Progress Report of the IDO Health and Safety Division (AEC no date) contains some facility environmental monitoring data. INEEL developed a routine facility environmental monitoring program between 1963 and 1970. In 1968 and 1969, formal Environmental Monitoring Reports (EMRs) (AEC '1968', AEC '1969') reported alpha, beta, and I-131 concentrations that can be correlated with the data of Table 4-1. The 1970 EMR (AEC '1970') reports gross beta values measured at the Central Facilities Area (CFA) that can be correlated with Table 4-1 (EBR-I) values. The analysis for this TBD reviewed EMRs between 1970 and 1990 for data that is available and usable for this correlation.

Table 4-2 lists results of the comparison. Because of the large variation in measurements made, the ratio of values calculated from the EMRs (column 4) to that derived from releases shown in Table 4-1 is not well behaved (geometric mean of 0.39 and geometric standard deviation of 5.6). Nevertheless this comparison provides confidence in the results of Table 4-1. The deviation of the average ratio from 1 is small compared to the default geometric standard deviation of 3 assumed for environmental results.

Table 4-2. Comparison of calculated facility intakes with intakes from environmental monitoring results .

Year	Activity type	Average annual Concentration	Annual inhaled quantity (Bq)	Table 4 inhaled quantity (Bq)
1963	β-γ	1.7E-11 μCi/cc	1510	1,310[a]
	Pu-239	6.0E-16 μCi/cc	0.05	0.014[a]
1968	α	0.0022 pCi/m^3	0.18	0.01[a]
	β	0.64 pCi/m^3	56	337[a]
	I-131	<0.08 pCi/m^3	<7.1	1.4[a]
1969	α	0.023 pCi/m^3	2	2.4E-3[a]
	β	2.95 pCi/m^3	262	118[a]
	I-131	0.123 pCi/m^3	10.9	2.1[a]
1970	Gross β	6.0E-13 μCi/ml	53	74[a]
	Max. gr. B @ CFA	8.1E-13 μCi/ml	72	74[a]
1973	Gross β	95 ±42 fCi/m^3	8.4	0.8[a]
EBR-I	Sr-90	3.4 fCi/m^3	0.3	0.15[b]
	Nb-95	1.0 fCi/m^3	0.09	---
	Cs-137	7-17 fCi/m^3	0.6-1.5	---
	Ce-144	4-8 fCi/m^3	0.36-0.71	0.057[b]
EFS	Sr-90	5.9 fCi/m^3	0.52	0.15[a]
	Nb-95	0.9-2.4 fCi/m^3	0.08-0.2	---
	Ru-106	6-9.8 fCi/m^3	0.53-0.87	0.27[a]
	Cs-134	0.8-1.6 fCi/m^3	0.07-0.14	---
	Cs-137	17-27 fCi/m^3	1.5-2.4	---
1976[c]	Gross β	3-6E-14 μCi/ml	2.5-5	0.6-25[d]
1986	Kr-85 @ CFA	3.7E-11 μCi/ml	3,290	890[e]
1988	Kr-85 @ CFA	1.1E-10 μCi/ml	9,770	14,000[e]
1990	Kr-85 @ CFA	2.7E-11 μCi/ml	2,400	690[e]

a. Values from INEEL TBD Table 4-3 for CFA.
b. Values from INEEL TBD Table 4-5 for RWMC since EBR-I is near RWMC.
c. Of 90 monthly values (January through September) for 10 facility areas, 89 values were between 3×10^{-14} and 6×10^{-14} μCi/ml.
d. Using the current tables with 11 radionuclides, the inhaled quantity is about 0.6 Bq; with the original tables with 44 radionuclides, the inhaled quantity is about 25 Bq.
e. Values derived from tables in an earlier version of the TBD report that contained concentrations of all INEEL released radionuclides.

Figure 4-3 shows the variation of INEEL Environmental Monitoring sampling results for the period from 1978 through 1986 which is typical for earlier years as well as for later years. This figure also shows the close correlation of environmental sample results acquired at “distant communities” and those acquired at INEEL facilities and the effect of foreign nuclear tests and the Chernobyl reactor accident on INEEL environmental sampling results. As shown on this figure, the INEEL average concentration has not differed from “distant community” concentrations by more than a factor of 2 for the 9-year period and is very similar for earlier and later periods; subsequent Environmental Monitoring Reports show the same correlation for the years before 1978 and after 1986. The greater perturbations in facility and distant community concentrations are nearly all correlated with fallout from nuclear tests. There was no discernible evidence that facility effluent or resuspension affected facility concentrations.

4.2.2 Episodic Releases at INEEL

Of the 108 episodic releases (accidents and planned tests) analyzed in DOE (1991), only 16 had the potential to affect other INEEL facilities. Only 9 of the 16 events could have affected the EBR-I facility. Section 4.1.2.2 describes three of these events, two of the three criticalities at ICPP and the Fuel Element Cutting Facility Filter Break, which also occurred at ICPP. The other 6 were planned GE-ANP tests that could have affected the EBR-I facility, as follows:

1. Initial Engine Test (IET) 14
2. IET 15(B)
3. IET 17(B)
4. IET 19(A)
5. IET 25(A)
6. IET 26(A)

For a given test, if there was an onsite facility between the point at which the test occurred and the affected offsite location, that test was conservatively assumed to have affected an onsite facility or facilities. For example, the FECF Filter Break occurred at ICPP and clearly contaminated an area south of ICPP. According to the meteorological dispersion at the time of the filter break, the affected offsite location was Frenchman’s Cabin. Because EBR-I is in the straight-line path between ICPP and Frenchman’s Cabin, a radiological impact analysis was conducted for the EBR-I.

Because all other test releases listed above, which originated at the TAN facility, affected one location on the southern boundary [Frenchman’s Cabin (shown on Figure 4-1), as evaluated in DOE (1991)], they have been assumed to have affected EBR-I because that facility is on the plume trajectory from TAN. The following sections discuss these events. The SL-1 accident, which was widely publicized, is only included to show it did not affect any other facility at the INEEL.

A concerted effort has been made to reduce the number of radionuclides involved in the releases for the episodic events. Overall, the mix of radionuclides for all the episodic events is complicated by the type (“aged” versus “fresh”), and the relative quantities of each. When viewed together, the episodic events can be categorized into three categories: criticalities that involve “fresh” fission products that have relatively short half-lives when compared to radionuclides released from the Fuel Element Burn Tests, for example, releases involving long half-lived, aged fission products (Fuel Element Burn Tests and the release from the Fuel Element Cutting Facility Filter Break), and releases from the remaining IET tests that released short half-lived radionuclides, which are generally characterized as “fresh” fission products, and long half-lived radionuclides, which are characterized as “aged” fission products. The latter category is unique to the GE-ANP Program because of the “direct-to-air” conversion nature of the tests. Therefore, within these categories, the number of radionuclides has been reduced to the number that preserves 95% of the original dose that was calculated for that particular location.

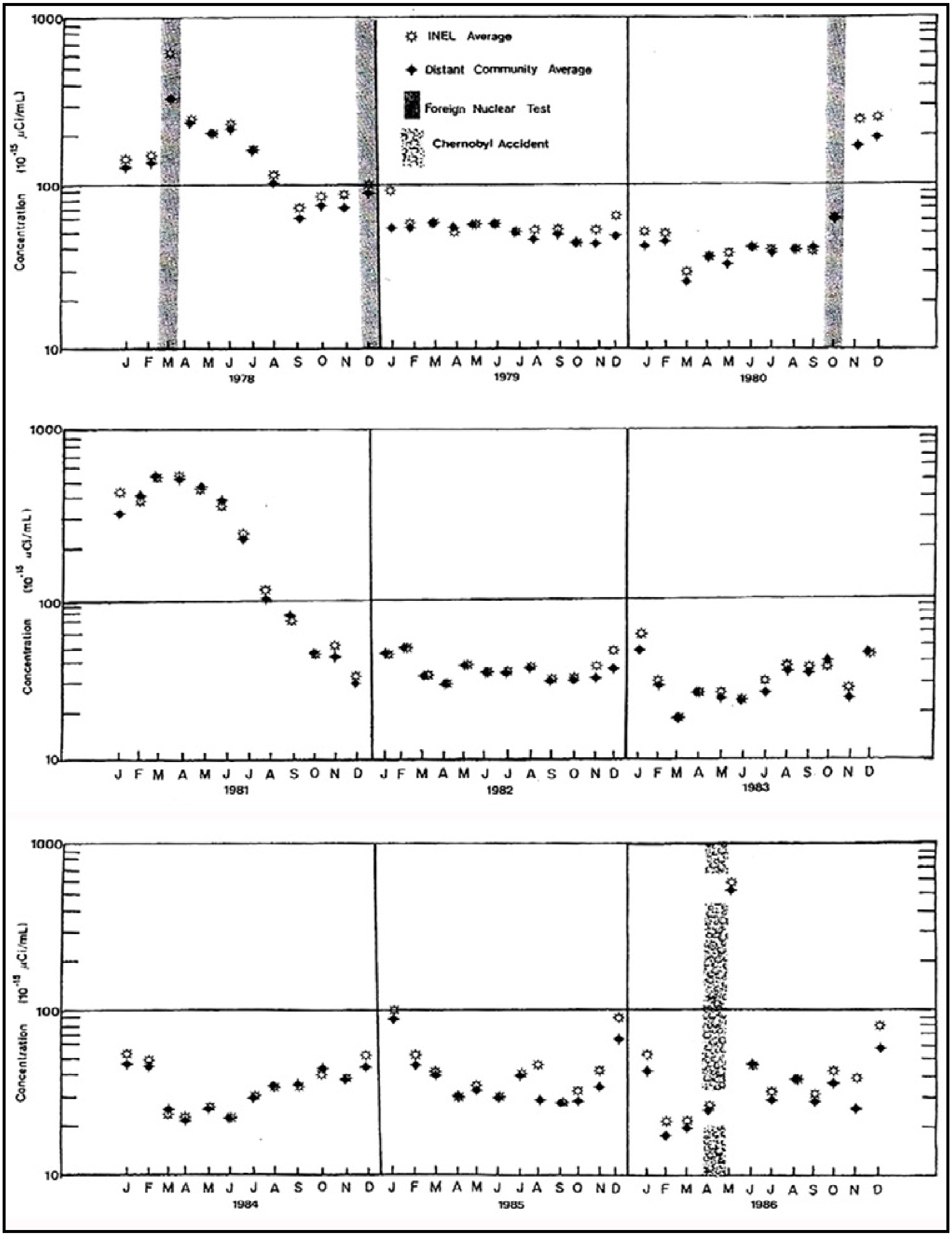

Figure 4-3. Onsite and distant particulate beta concentrations in air.

4.2.2.1 SL-1 Reactor Accident

One significant accident at INEEL in the last 51 years released substantial amounts of RM to the environment. On January 3, 1961, a steam explosion at the Stationary Low-Power Reactor No. 1 (SL-1) facility [near the location of Auxiliary Reactor Area (ARA) II in Figure 4-1] killed three SL-1 personnel and ruptured the SL-1 reactor vessel. This, in turn, propelled RM into the reactor building and then into the environment. The amount of the release and the path that the cloud traveled from the reactor building were carefully monitored and well documented (Gammill 1961; Horan and Gammill 1961; Kunze 1962). All radiological doses to personnel involved in the rescue and cleanup of the reactor building were carefully controlled and documented.

The SL-1 accident did not affect any other INEEL facility with the effluent of RM. The effluent traveled to the south of the facility, as shown in Figure 4-4 (DOE 1991).

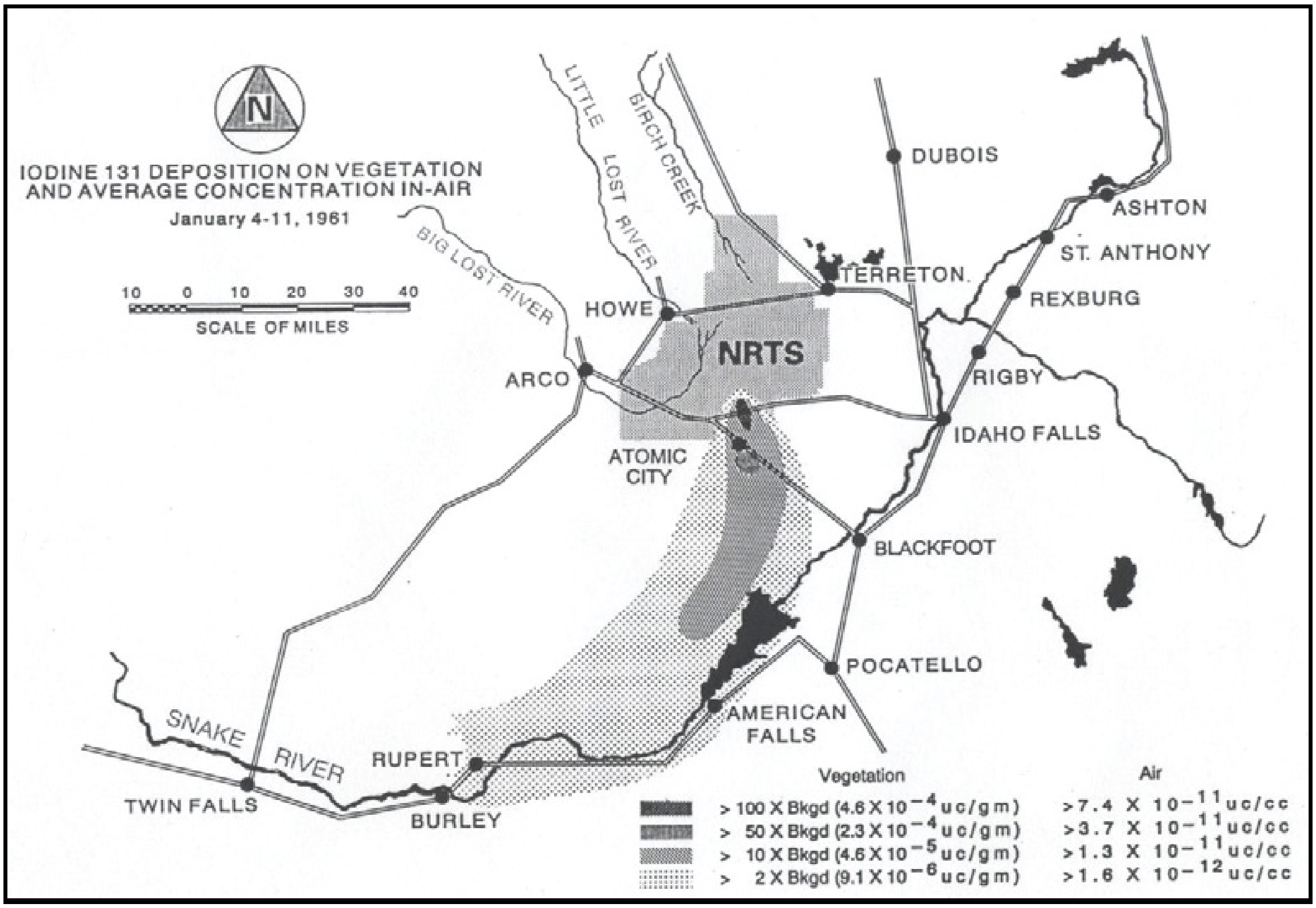

Figure 4-4. Dispersion coefficient contours for SL-1 accident (redrafted from Horan and Gammill 1961).

4.2.2.2 Criticality and Accident Occurrences at ICPP

Three accidental criticalities have occurred at the ICPP (now INTEC). The first occurred on October 16, 1959, the second on January 25, 1961, and the third on October 17, 1978. The 1978 criticality, which released essentially just noble fission gases produced during the criticality, occurred after ANL-W operations had moved to the present South-East site location. The two earlier criticalities released RM during or shortly after the event; in both cases, the effluent was transported to the south-southwest and potentially exposed personnel at the EBR-I facility. The Fuel Element Cutting Facility Filter Break accident also occurred at the ICPP and has been postulated to have impacted the EBR-I facility. Claimant favorable analyses, described below, defined the amount of potential radiological exposure that could have occurred to an individual at this location.

4.2.2.2.1 ICPP Criticality of October 16, 1959

On October 16, 1959, at approximately 3:00 a.m., a criticality event occurred at the ICPP in the WH-100 vessel. The estimated magnitude of this event was no greater than 4×10^{19} fissions (DOE 1991). *Nuclear Incident at the Idaho Chemical Processing Plant* (Ginkel et al. 1960) provides a full account of the incident and documents the radiological doses, calculated internal and measured external, for plant personnel involved in the incident.

For the calculation of intakes for this incident, meteorological conditions were modeled so the X/Q at 22 km matched the value calculated for Frenchman's Cabin (south of the INEEL, as shown on Figure 4-1) where offsite doses were calculated and reported in DOE (1991). RSAC-6 was used to calculate X/Q values for EBR-I. These concentrations and intake quantities would be applicable only if the individual was in the respective areas on the morning of October 16, 1959. Table 4-3 lists intakes applicable at EBR-I.

Table 4-3: Intakes (Bq/event) at EBR-I for criticalities at ICPP.

Date	**10/16/1959**	**1/25/1961**
Event	**Criticality**	**Criticality**
Exposure location	**EBR-I Area**	**EBR-I Area**
Rb-89	2.1E+4	4.4E-1
Sr-91	2.2E+3	4.8E+0
Sr-92	2.6E+3	4.3E+0
Y-92	3.1E+2	2.2E+0
Y-93	2.4E+2	5.7E-1
Te-133		
I-131(elem.)	1.2E+1	5.7E-2
I-133	2.6E+2	1.1E+0
I-134	1.9E+3	2.1E+0
I-135	8.2E+2	3.2E+0
Cs-138	3.1E+4	
Ba-139	1.6E+4	1.9E+1
La-141	1.1E+3	3.0E+0
La-142	9.4E+2	1.2E+0

4.2.2.2.2 ICPP Criticality of January 25, 1961

The January 25, 1961 criticality occurred in ICPP vessel H-110 about 9:50 a.m. This event consisted of an estimated 6.0×10^{17} fissions. The report documenting the incident states:

> *Of the 251 individuals present in the ICPP area at the time of the incident, none received significant radiation exposure. The highest exposure as determined from film badge readings did not exceed 55 millirem of penetrating radiation. Essentially no beta radiation was detected. No significant neutron exposure or internal contamination from inhalation was found. The absence of significant exposures is attributable to the extensive shielding provided by the process cell in which the event took place and the control of the fission gases by the equipment. (Paulus et al 1961)*

As for the 1959 criticality, X/Q values were calculated for EBR-I. The source term used for this event is the same as that used for DOE (1991). Table 4-3 lists the intakes applicable at EBR-I if the individual was in the area on January 25, 1961.

4.2.2.2.3 Fuel Element Cutting Facility Filter Break

The ends of fuel elements sent to ICPP contained structural components that were cut off before the elements were processed. Cutting these end pieces off and cutting the fuel elements into sections before they were sent to CPP-601 for processing occurred in the FECF in CPP-603. During the night of October 29 and early in the morning of October 30, 1958, INEEL conducted decontamination operations in the FECF. Acid fumes from the decontamination operations caused failure of the FECF exhaust filters, resulting in the release of particulate activity to the south of ICPP.

Approximately 100 curies (Ci) of long half-life particulate RM was released over an area of approximately 200 acres (AEC 1959). The released radioactive material and quantities were the same as those published in DOE (1991). Table 4-4 lists the best-estimate intakes of the radionuclides in the EBR-I area. These intakes would be applicable only if the individual was in the area at the time of the release (i.e., during the night of October 29 and the early morning of October 30, 1958).

Table 4-4. Intakes (Bq/event) at EBR-I for FECF filter break incident at ICPP .

Date	Event	Exposure location	Sr-89	Sr-90	Y-91	Zr-95	Ru-103	Ru-106	I-131 (elem.)	Ce-144	Pr-143
10/29/58 10/30/58	**FECF Filter Break**	**EBR-I Area**	6.3E-1	6.3E-1	1.4E+0	2.0E+0	1.3E-1	6.6E-1	4.2E-12	9.0E+0	2.6E-6

4.2.2.3 Releases from Planned Tests

The following were all planned tests, conducted under the GE-ANP Program at Test Area North (TAN), which potentially affected the EBR-I facility.

4.2.2.3.1 Initial Engine Test 14

IET 14 was the eighth nuclear test conducted by the GE-ANP program at TAN. This test was the fifth in the HTRE-2 reactor configuration. This test series involved the evaluation of the L2A-1 insert cartridge. The cartridge contained fueled and unfueled, beryllium-oxygen ceramic tubes. There was no coating on the inside surfaces of the fueled tubes (Pincock 1959).

A total of 100.25 hours was accumulated on the insert fuel cartridge at a maximum insert fuel temperature of approximately 2,500°F. The objectives of the test were to (1) evaluate the operational effect of water vapor corrosion on fueled beryllium-oxygen tubes operating at a constant reactor mixed mean discharge air temperature over approximately 100 hours, and (2) measure the fission product release rate from uncoated fueled tubes as a function of temperature and operating time (Pincock 1959).

Table 4-5 lists the fission products released during the IET 14 test, and the intakes at EBR-I. An individual would have been exposed to these concentrations only if present at these locations between April 24 and May 19, 1959. The above intakes are for a total exposure period of 26 days.

Table 4-5: Intakes (Bq/event) at EBR-I for Initial Engine Tests at INEEL .

Period	Test	Exposure location	Rb-89	Sr-89	I-131 (elem.)	I-133	I-135	Cs-138	U-234
4/24 – 5/19/59	IET 14	EBR-I	5.8E+0	1.7E-2	2.1E+0	1.3E+1	1.8E+1	4.2E+1	3.6E-6
6/16-6/24/59	IET 15(B)	EBR-I	3.3E-1	8.4E-4	4.5E-1	2.0E+0	3.2E+0	2.3E+0	3.9E-5
10/12-12/12/59	IET 17(B)	EBR-I	2.2E-3	6.2E-3	5.7E-1	2.4E+0	2.5E+0	6.2E-1	5.5E-6
2/17 – 2/29/60	IET 19(A)	EBR-I	1.3E-1	6.8E-3	9.9E-1	6.7E+0	9.9E+0	4.9E+0	4.2E-7
11/22-11/30/60	IET 25(A)	EBR-I	7.1E-4	4.6E-4	3.6E-1	4.0E+0	5.1E+0	1.5E-1	1.5E-6
12/23-12/28/60	IET 26(A)	EBR-I	1.7E+0	1.1E-2	1.8E+0	7.2E+0	1.1E+1	1.5E+1	2.5E-5

4.2.2.3.2 Initial Engine Test 15(B)

IET 15 was conducted at TAN between May 27 and June 24, 1959. This test involved the evaluation of the L2C-1 insert cartridge, which was of the concentric ring design. The fuel sheet was made of a chromium-uranium dioxide-titanium core clad with an iron-chromium-yttrium alloy (Evans 1959). From this operation data was obtained to evaluate:

1. Endurance capabilities of the advanced metals at a design temperature of 2,000°F for extended periods (planned endurance testing to total 120 hours or more)

2. The structural and metallurgical integrity of the fuel sheet in this particular cartridge design

3. The nature and extent of fuel sheet damage, if any, and the effect on cartridge performance

4. The performance potentials of the cartridge

The operation was successfully conducted to accumulate 80.75 hours at an insert extrapolated fuel sheet temperature of 2,015°F. The operation was terminated after 80.75 hours due to a release of fission products of such a quantity as to indicate fuel sheet rupture of an extent sufficient to warrant inspection (Evans 1959).

The insert was visually examined after completion of testing. No damage had occurred to the outer fuel sheets of the cartridge; however, there were blisters on the inner fuel sheets. In some instances the blisters had ruptured. The FP release for this test was divided into two periods based on a review of effluent monitoring data. The period from June 3 to 15 was considered to be an operation before the development of significant fuel sheet blisters. June 16 to 24 comprised the second period, when effects of blistering were clearly observed.

According to the meteorology of the testing period, the second period affected Frenchman's Cabin. Accordingly, this analysis addressed the radiological impact on the EBR-I. Table 4-5 lists the release of fission products, which correspond with the Part B operation release documented in DOE (1991) for the intakes applicable at EBR-I. An individual would have been exposed to these concentrations and intakes only if present at the locations between June 16 and June 24, 1959. The above intakes are for the total 9-day exposure period.

4.2.2.3.3 Initial Engine Test 17(B)

IET 17 occurred between October 12 and December 12, 1959. Releases of airborne radioactivity occurred between November 2 and December 12, 1959, when the reactor operated at power levels exceeding 100 kW. The test series involved the evaluation of the L2E-1 insert cartridge (Evans 1960). Table 4-5 lists the intakes for an individual at EBR-I. An individual would have received these intakes only if present at the respective locations between November 2 and December 12, 1959. The intakes are for a total exposure period of 40 days.

4.2.2.3.4 Initial Engine Test 19(A)

IET 19, conducted between February 9 and April 30, 1960, was a test series in the HTRE No. 2 reactor to evaluate the L2E-3 insert, which contained fueled and unfueled hexagonal beryllium-oxygen ceramic tubes. The tubes were coated on the inside with coextruded zirconia (zirconium dioxide) (Pincock 1960). The primary purposes for running the test were to:

1. *Operate the L2E-3 fuel cartridge at peak temperatures of 2,500° F and 2,600° F for 100 hours or more at each temperature level to evaluate the effectiveness of the zirconium-dioxide coating against hydrolysis and the release of fission products.*

2. *Operate the insert fuel cartridge at various temperature levels at specified intervals during the endurance testing to determine fission product release as a function of insert temperature.*

3. *Obtain additional information on the effectiveness of an electrostatic precipitator in removing fission products from the reactor effluent. (Pincock 1960).*

Pincock (1960) summarized the estimated total fission product release for the test runs based on spot sampling and reported them as 10-minute-decayed curies. The total fission product release reported for IET 19 was 2,892 Ci. The release for this test was modeled as for DOE (1991). Table 4-5 lists intakes for EBR-I. An individual would have been exposed to these intakes only if present at the respective locations between February 17 and February 29, 1960. The intakes are for the total exposure period of 13 days.

4.2.2.3.5 Initial Engine Test 25(A)

IET 25, performed between November 15 and December 19, 1960, was an extension of the Phase II testing program conducted in IET 18. The test was conducted in the HTRE No. 3 reactor configuration. Releases of airborne radioactivity corresponding to the significant periods of operation were assumed to have occurred between November 22 and December 15, 1960. The release at IET 25(A) was assumed to have occurred from November 22 through November 30, 1960.

> *The purposes of test series IET No. 25 were to demonstrate the capabilities of the fuel elements above design temperatures and to confirm that the powerplant could achieve a full nuclear start as predicted. The reactor went critical on November 14, 1960, and the test program was completed on December 19, 1960. (Linn 1962).*

Only the following summary of effluent monitoring activities and results was available:

> *Continuous effluent monitoring was maintained to measure and record the activity released to the atmosphere by the powerplant. The maximum output was 3.4 curies/hour (measured 10 minutes after release). The total output for the test series was 218 curies (measured 10 minutes after release). The maximum release rate for I-131 was approximately 0.7 curies/hour (measured 10 minutes after release). The total offsite inhaled and ingested dose was below measurable amounts during this test series. (Highberg et al. 1961)*

For this analysis the release was modeled, as in DOE (1991), as a straight-line trajectory such that the centerline plume affected EBR-I. Table 4-5 lists the intakes for this test. An individual would have received intakes only if present at the respective locations between November 22 and December 15, 1960. The tabulated intakes are for a total exposure period of 24 days.

4.2.2.3.6 Initial Engine Test 26(A)

IET 26, conducted in HTRE No. 2, occurred between December 22, 1960, and March 31, 1961 (Field 1961). Releases of airborne activity for the total test were assumed to have occurred between December 23, 1960, and March 30, 1961, when the reactor operated at power levels exceeding 120 kW. Releases for the IET 26(A) operation occurred from December 23 to 28, 1960. The insert under test was the L2E-6 cartridge, which consisted of fueled and nonfueled ceramic beryllium-oxide hexagonal tubes coated on the inner surface with zirconium dioxide.

The airborne release model was consistent with the model of DOE (1991) with an assumed straight-line trajectory between TAN and EBR-I. Table 4-5 lists the intakes for EBR-I. An individual would have been exposed to these intakes only if present at the respective locations between December 23 and December 28, 1960. The tabulated intakes are for a total exposure period of 6 days.

4.3 EXTERNAL DOSE

External radiation dose at a facility can be created by direct radiation from two sources: direct beta/gamma radiation from the facility or airborne effluents released from the facility or from adjacent facilities. In general, direct beta/gamma radiation from the facility will increase with time because the general contamination of the area will increase. In addition, as a facility ages, radioactive sources tend to accumulate at the facility, causing the general background to increase with time. A responsible H&S organization will observe and curb such a trend to prevent personnel exposures from increasing unnecessarily. The following sections discuss facility fence-line film badge and TLD data that recorded doses from airborne fission product releases that had the potential for personnel exposure. Peterson (2004) contains more information on these two subjects.

4.3.1 Facility Fence-Line Annual Doses

Before 1970, many film badge or TLD measurements occurred inside the INEEL Site boundary. During the IET period at TAN (1956 to 1961), many film badges were placed along the highways that triangulated the IET area and along some of the highways at the southern end of the Site. Initially, the badges were retrieved and read once a month. The frequency changed to 6 weeks in 1962 and then changed back to monthly in 1963. Film badges were used through 9 months of 1966 and TLDs were used after that time. Beginning in 1967, TLDs were changed on a semiannual basis. Recorded significant readings during the film badge period showed that the maximum badge reading increased by only a factor of 2 or 3 above background. However, the location of the badge with the increased reading was not identified. Peterson (2004) contains more information and film badge data for this early period. The “detection limit” for the film badge reading was often quoted as 10 mrem for both beta and gamma readings (AEC 1963) and as 10 mrem for the TLD when it was first used. With the measured annual background radiation field at INEEL before operations began between 100 and 150 mrem/year, the monthly value of 8 to 13 mrem is at the detection limit of the film badge or TLD. Therefore, the uncertainty for monthly changeouts is higher than for less frequent changeouts.

Facility fence monitoring and facility locations were established between the latter part of 1970 and the latter part of 1972. No film badge or TLD information is available for the EBR I and BORAX locations when they were operational. From 1972 through 1983, facility fence TLD measurements, made on a 6-month basis with 5 TLDs at each facility position, are available in the Environmental

Monitoring Data Reports (EMDRs) for INEL. Figure 4-5 (Table II from ERDA 1976) shows that uncertainty can vary from less than 10% to 20% for a given set of readings. At each of the 34 listed monitoring locations, there were normally five TLDs for a potential of 170 readings for a 6-month period. For this particular 2-year set of data, 1.5% of the 136 sets of readings are assigned a 2 sigma uncertainty of 16% to 20%, and 18.4% of the readings are assigned a 2 sigma uncertainty of 11% to 15%. However, 80% of the 136 values ascribed for the 34 locations over the 2-year period have a 2 sigma uncertainty of less than 10%.

TABLE II

ONSITE PENETRATING RADIATION EXPOSURE DATA

Facility	Badge Location Number	Adjusted Six-Month Exposure, mR* 5/74-10/74	11/74-4/75	5/75-10/75	11/75-5/76
ARA-I & II	1	120[a]	100[a]	121	101
	2	200	100	138	112
	3	100[a]	260[a]	82	70
	4	1750[a]	670	262	200
SPERT-PBF	1	74	65[a]	68[a]	90
	2	71	64	66	61[a]
	3	68	61	66[a]	65[a]
	4	74	64	78	65
	5	70	67	70[a]	64
	6	71	65	71	71
TAN-TSF	1	65	75[b]	72	64
TAN-LOFT	2	67	66[a]	65	62
	3	68	73	70	69
	4	58	57	56	53
TAN-LPT	5	62	63	65	64[a]
	6	62	58	61	57
	7	60	60	62	60
	8	65	62	66	58
CFA	1	72	65	68	67
	2	70	65	72	66
	3	65	66	69	63
TRA	1	130	133[a]	111	88[a]
	2	200	170	166	120[a]
	3	1000[a]	810	659	540
	4	1500[a]	1080	1133	1010
	5	2460	1890	2434	2100
	6	1950[a]	1870	664	96
	7	280[a]	280[a]	274	250
	8	530	500[a]	538[a]	500
	9	290	270	269	230
	10	86	85[a]	93	83
	11	83	77	85	83
	12	100[b]	100	86	85
	13	190	160[a]	108	82

* - 2 sigma was 10% or less except where noted.
a - 2 sigma was 11 to 15%.
b - 2 sigma was 16 to 20%.

Figure 4-5 Example of onsite TLD monitoring data.

To supply facility values for the 1965-1972 period, the highest value from all subsequent years was used. Facility fence TLD measurements could not be located for 1984 through 1992, but for 1993 and beyond the INEL/INEEL EMRs include such facility fence-line measurements. For the 1984-1992 period for which TLD measurements are missing, reasonable extrapolations provided the missing values. In addition, the EMRs recorded background TLD measurements corresponding to the facility fence TLD measuring periods. Table 4-6 lists all reduced facility fence-line TLD data (facility fence-line data minus background) in the EMRs. Peterson (2004) contains a more detailed discussion of the data.

Table 4-6. INEEL facility fence direct gamma values (TLD – background) (mR).

Year	EBR-II	TREAT	Background
1965-72	59	50	100-150
1973	37	19	121
1974	35	17	123
1975	32	8	118
1976	56	50	113
1977	22	0	132
1978	56	2	129
1979	59	5	113
1980	51	12	119
1981	28	9	118
1982	20	12	117
1983	24	10	115
1984	31	13	124
1985	31	13	124
1986	31	13	124
1987	31	13	124
1988	31	13	124
1989	31	13	124
1990	19	13	124
1991	19	13	124
1992	19	13	124
1993	28	16	111
1994	15	3	130
1995	17	7	116
1996	22	21	129
1997	16	16	128
1998	0	11	131
1999	13	13	122
2000	25	26	129
2001	0	3	140
2002	18	39	120

4.3.2 Facility Air Immersion Doses

INEEL facility air-immersion (beta-gamma) doses could be calculated from the noble gas and halogen portions of the operational releases, and, if applicable, from the noble gas portion of the applicable episodic releases. This calculation should be unnecessary because these releases would be recorded in the fence-line TLD doses listed in Table 4-6.

However, in considering this increased uncertainty, it is interesting to note facility air monitoring results that are also discussed in the annual EMRs. In each case, the facility air concentration is compared to that concentration for a distant community, usually Idaho Falls. Normally the concentration is indistinguishable from the concentration for the distant community, as discussed in Section 4.2.1.

4.4 UNCERTAINTY

INELHDE (DOE 1991) contains a detailed discussion of the derivation of airborne releases for operational conditions and episodic events.

Operational Releases

Discussions with the INELHDE (DOE 1991) authors suggest that operational releases, which were monitored, could be low by a factor of not more than 2. When the annual normalized ground-level concentration values are applied to the operational releases, the uncertainty could increase.

Episodic Releases

As described in INELHDE (DOE 1991), the episodic releases are a maximum reasonable value, based on the amount of material available to be released, and the conditions of the respective test. For such a release, the inhaled quantities (in Bq) were maximized by assuming the downwind exposed individual was subjected to the plume centerline concentration for the total time, night and day in most cases, of the release. In spite of the original effort to be “reasonably conservative” in the exposure estimates, some of the authors stated that the release considered for a particular episodic event might be low by as much as a factor of 3.

Film Badge and TLD Measurements

As discussed in Section 4.3.1, the uncertainty of individual measurements, made with film badges and TLDs, can be as high as ±100%, depending on the frequency of changeout (i.e., once per month, which was generally the case with film badges). The data for 1965 to 1972 in Table 4-6 is based on the highest TLD 6-month values of 1972 for the respective facility. Although the GE-ANP IET tests were conducted in the late 1950s and early 1960s (the last IET, 26, ended on March 31, 1961), tests with planned releases were administratively and meteorologically controlled so the airborne effluent traveled to the northeast over the monitoring grid such that adjacent facilities were not affected. However, after 1967 when facility fence-line measurements were routinely made with TLDs, with five TLDs at a given location the uncertainty is generally ascribed at less than 10%. Occasionally, less than about 20% of the time, these measurements have an ascribed level of uncertainty as high as 20%.

REFERENCES

AEC (U.S. Atomic Energy Commission), 1959, *1958 Annual Report of the Health and Safety Division,* IDO-12012, Idaho Operations Office, Idaho Falls, Idaho.

AEC (U.S. Atomic Energy Commission), 1963, *Annual Progress Report 1962, Health & Safety Division,* IDO-12033, Idaho Operations Office, Idaho Falls, Idaho.

AEC (U.S. Atomic Energy Commission), no date, *Annual Progress Report 1963, Health & Safety Division,* IDO-12037, Idaho Operations Office, Idaho Falls, Idaho.

AEC (U.S. Atomic Energy Commission), '1968', *Environmental Monitoring Report No. 23, July – December and Annual Summary – 1968,* Idaho Operations Office, Health Services Laboratory, National Reactor Testing Station.

AEC (U.S. Atomic Energy Commission), '1969', *Environmental Monitoring Report No. 25, July – December and Annual Summary – 1969,* Idaho Operations Office, Health Services Laboratory, National Reactor Testing Station.

AEC (U.S. Atomic Energy Commission), '1970', *National Reactor Testing Station Environmental Monitoring Program Report, January – December 1970,* Idaho Operations Office, Health Services Laboratory, Idaho Falls, Idaho.

Bowman, A. L., 1984, *INEL Environmental Site Characterization Report,* EGG-NPR-6688, 3 volumes.

Casto, W. R., 1980, "ICPP Criticality Event of October 17, 1978," *Nuclear Safety*, 21, 5.

DOE (U.S. Department of Energy), 1979, *1978 Environmental Monitoring Program Report for the Idaho National Engineering Laboratory Site*, IDO-12082(78), Idaho Operations Office, Idaho Falls, Idaho.

DOE (U.S. Department of Energy), 1989, *The Idaho National Engineering Laboratory Site Environmental Report for Calendar Year 1988*, IDO-12082(88), Idaho Operations Office, Idaho Falls, Idaho.

DOE (U.S. Department of Energy), 1991, *Idaho National Engineering Laboratory Historical Dose Evaluation*, DOE/ID-12119, Volumes 1 and 2, Idaho Operations Office, Idaho Falls, Idaho.

ERDA (Energy Research and Development Administration), 1976, *INEL Site Environmental Monitoring Data-First Quarter 1976,* Idaho Operations Office, September

ESRF (Environmental Science and Research Foundation) 1994, *The Idaho National Engineering Laboratory Site Environmental Report for Calendar Year 1993,* DOE/ID-12082(93), Idaho Falls, Idaho.

ESRF (Environmental Science and Research Foundation) 1995, *The Idaho National Engineering Laboratory Site Environmental Report for Calendar Year 1994,* DOE/ID-12082(94), Idaho Falls, Idaho.

ESRF (Environmental Science and Research Foundation) 1996, *In Summary: Idaho National Engineering Laboratory Site Environmental Report for Calendar Year 1995,* DOE/ID-12082(95), Idaho Falls, Idaho.

ESRF (Environmental Science and Research Foundation, Inc.) 1997, *Idaho National Engineering and Environmental Laboratory Site Environmental Report for Calendar Year 1996,* DOE/ID-12082(96), Idaho Falls, Idaho.

ESRF (Environmental Science and Research Foundation, Inc.) 1998, *Idaho National Engineering and Environmental Laboratory Site Environmental Report for Calendar Year 1997,* DOE/ID-12082(97), Idaho Falls, Idaho.

ESRF (Environmental Science and Research Foundation, Inc.) 2000, *Idaho National Engineering and Environmental Laboratory Site Environmental Report for Calendar Year 1998,* DOE/ID-12082(98), Idaho Falls, Idaho.

ESRF (Environmental Science and Research Foundation, Inc.) 2001, *Idaho National Engineering and Environmental Laboratory Site Environmental Report for Calendar Year 1999,* DOE/ID-12082(99), Idaho Falls, Idaho.

ESRF (Environmental Science and Research Foundation, Inc.) 2002, *Idaho National Engineering and Environmental Laboratory Site Environmental Report for Calendar Year 2000,* DOE/ID-12082(00), Idaho Falls, Idaho.

ESRF (Environmental Science and Research Foundation, Inc.) 2002, *Idaho National Engineering and Environmental Laboratory Site Environmental Report for Calendar Year 2001,* DOE/ID-12082(01), Idaho Falls, Idaho.

ESRF (Environmental Science and Research Foundation, Inc.) 2002, *Idaho National Engineering and Environmental Laboratory Site Environmental Report for Calendar Year 2002,* DOE/ID-12082(02), Idaho Falls, Idaho.

Evans, R. C., 1959, *Operations Report, HTRE-2, L2C-1 Cartridge*, XDC 59-11-179, November 16.

Evans, R. C., 1960, *HTRE-2 Insert L2E-1 Cartridge Operations Report*, XDC 60-3-111, March 8.

Field, J. H. (editor), 1961, *Test Results from D101-L2E-6 Power Testing, IET #26*, DC 61-6-704.

Gammill, W. P., 1961, "Environmental Aspects of the SL-1 Accident," *7th Annual Meeting of the Pacific Northwest Section of the American Industrial Hygiene Association*, September 22, 1961.

Ginkel, W. L., C. W. Bills, A. O. Dodd, K. K. Kennedy, and F. H. Tingey, 1960, *Nuclear Incident at the Idaho Chemical Processing Plant*, IDO-10035, U.S. Department of Energy, Idaho Operations Office, Idaho Falls, Idaho.

Highberg, J. W., J. H. Field, K. J. Skow, C. B. Cannon, and C. Davidson, 1961, *Preliminary Data Report IET Test Series No. 25 D102A2 Powerplant*, DC 61-2-724, February 10.

Horan, J. R., and W. P. Gammill, 1961, "The Health Physics Aspects of the SL-1 Accident," *Health Physics Society Meeting*, Las Vegas, Nevada, June 16, 1961.

Kunze, J. F., 1962, *Additional Analysis of the SL-1 Excursion, Final Report of Progress, July through October 1962,* IDO-19313, U.S. Department of Energy, Idaho Operations Office, Idaho Falls, Idaho.

Linn, F. C., 1962, *Heat Transfer Reactor Experiment No. 3*, *General Electric Direct-Air-Cycle Aircraft Nuclear Propulsion Program*, APEX-906, June 15.

ORAU (Oak Ridge Associated Universities), 2004, *Technical Basis Document for the Argonne National Laboratory-West (ANL-W) – Site Description*, ORAUT-TKBS-0026-2, Oak Ridge, Tennessee.

Paulus, R. C., A. O. Dodd, K. K. Kennedy, F. H. Tingey, and F. M. Warzel, 1961, *Nuclear Incident at the Idaho Chemical Processing Plant, January 25, 1961*, IDO-10036, U.S. Department of Energy, Idaho Operations Office, Idaho Falls, Idaho.

Pincock, G. D. (editor), 1959, *Test Results from D101 l2a-1 Power Testing, IET #14*, DC 59-8-728, August 19.

Pincock, G. D., 1960, *Test Results from D101-L2E-3 Power Testing, IET #19*, DC 60-7-717, July 8.

Peterson, H. K., 2004, *Supporting Information for INEEL Environmental Dose TBD*, February.

RAC (Radiological Assessment Corporation), 2002, *Final Report, Identification and Prioritization of Radionuclide Releases from the Idaho National Engineering and Environmental Laboratory*, Centers for Disease Control and Prevention, Department of Health and Human Services, Atlanta, Georgia.

Stoller (S. M. Stoller Corporation), 2002a, *Idaho National Engineering and Environmental Laboratory Site Environmental Report Calendar Year 1999,* DOE/ID-12082(99), Idaho Falls, Idaho.

Stoller (S. M. Stoller Corporation), 2002b, *Idaho National Engineering and Environmental Laboratory Site Environmental Report Calendar Year 2000,* DOE/ID-12082(00), Idaho Falls, Idaho.

Stoller (S. M. Stoller Corporation), 2002c, *Idaho National Engineering and Environmental Laboratory Site Environmental Report Calendar Year 2001,* DOE/ID-12082(01), Idaho Falls, Idaho.

Wenzel, D. R., 1990, *Interim User's Manual for RSAC-4, Radiological Safety Analysis Computer Program, Version 4.03*, Westinghouse Idaho Nuclear Company, Inc.

Wenzel, D. R., and B. J. Schrader, 2001, *The Radiological Safety Analysis Computer Program (RSAC-6) Users' Manual*, INEEL/EXT-01-00540, U.S. Department of Energy, Idaho National Engineering and Environmental Laboratory, Idaho Falls, Idaho.

Yanskey, G. R., E. H. Markee, Jr., and A. P. Richter, 1966, *Climatography of the National Reactor Testing Station,* IDO-12048, Idaho Operations Office, Idaho Falls, Idaho