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PUBLICATION RECORD

EFFECTIVE DATE	REVISION NUMBER	DESCRIPTION
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ACRONYMS AND ABBREVIATIONS

AEC	U.S. Atomic Energy Commission
AMAD	activity median aerodynamic diameter
ANL-W	Argonne National Laboratory–West
ARA	Auxiliary Reactor Area
ATR	Advanced (previously Army) Test Reactor
BORAX	Boiling Water Reactor Experiment
Bq	becquerel
CFA	Central Facilities Area
CFSGF	Coal-Fired Steam Generating Facility
CFR	Code of Federal Regulations
Ci	curie
CTF	Core Test Facility
d	day
DOE	U.S. Department of Energy
EBR	Experimental Breeder Reactor
ECF	Expended Core Facility
EEOICPA	Energy Employees Occupational Illness Compensation Program Act of 2000
EFS	Experimental Field Station
EMR	Environmental Monitoring Report
EOCR	Experimental Organic-Cooled Reactor
ETR	Engineering Test Reactor
F	fast (solubility rate)
FAST	Fluorinel Dissolution Process and Fuel Storage (facility)
fCi	femtocurie
FECF	Fuel Element Cutting Facility
ft	foot
GE-ANP	General Electric-Advanced Nuclear Propulsion (Program)
H&S	Health & Safety
hr	hour
HTRE	Heat Transfer Reactor Experiment
ICPP	Idaho Chemical Processing Plant
IDO	Idaho Operations Office
IET	Initial Engine Test
INEEL	Idaho National Engineering and Environmental Laboratory
INEL	Idaho National Engineering Laboratory
INELHDE	*Idaho National Engineering Laboratory Historical Dose Evaluation*
INL	Idaho National Laboratory
INTEC	Idaho Nuclear Technology and Engineering Center
km	kilometer
kW	kilowatt

m	meter
M	moderate (solubility rate)
mi	mile
mL	milliliter
mR	milliroentgen
mrem	millirem
MTR	Materials Test Reactor
NRF	Naval Reactors Facility
NCRP	National Council on Radiation Protection and Measurements
NIOSH	National Institute for Occupational Safety and Health
NOAA	National Oceanic and Atmospheric Administration
NRTS	National Reactor Testing Station
NWCF	New Waste Calcining Facility
ORAU	Oak Ridge Associated Universities
pCi	picocurie
POC	probability of causation
PREPP	Process Experimental Pilot Plant
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
SPERT	Special Power Excursion Reactor Test (facility)
SRDB Ref ID	Site Research Database Reference Identification
STSL	Semi Scale Support Laboratory
SWEPP	Stored Waste Examination Pilot Plant
TAN	Test Area North
TBD	technical basis document
TLD	thermoluminescent dosimeter
TRA	Test Reactor Area
TREAT	Transient Reactor Test
TSF	Technical Support Facility
U.S.C.	United States Code
WERF	Waste Experimental Production Facility
wk	week
WRRTF	Water Reactor Research Test Facility
yr	year
ZPPR	Zero Power Plutonium (later Physics) Reactor
α	alpha activity

β	beta activity
γ	gamma activity
µCi	microcurie
°F	degree Fahrenheit
§	section or sections

4.1 INTRODUCTION

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

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

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

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

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

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

4.1.1 Purpose

This TBD provides the basis for the environmental doses at the Argonne National Laboratory–West (ANL-W) facility at the Idaho National Laboratory (INL). This information can be used in dose reconstructions for the EEOICPA to supplement information in an individual's dose record.

This TBD addresses radioactive material (RM) releases from areas or facilities at INL, formerly the National Reactor Testing Station (NRTS) and then the Idaho National Engineering Laboratory (INEL) and the Idaho National Engineering and Environmental Laboratory (INEEL), that could affect employees at the 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 (DOE 1991). 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 from facility operations. In general, these doses, if not controlled by management, increase with time and create a *facility background dose*. At INL, these facility background doses were recorded by film badges infrequently and inconsistently before 1970 and by thermoluminescent dosimeters (TLDs) on a routine basis since 1970 (Walker 1971). These facility background doses, or *facility fenceline 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 fenceline doses (minus background) for the Environmental Breeder Reactor No. I (EBR-I) location for 1952 to 1972 and the Transient Reactor Test (TREAT) facility and EBR-II locations, both for 1972 to the present.

4.1.2 Scope

As outlined and discussed in Section 2.0 of the ANL-W site profile (ORAUT 2006), the U.S. Atomic Energy Commission (AEC) selected the INL site as an isolated location for the testing of reactor concepts. INL 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 INL 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 ft/d (ERDA 1977a).

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

During the 50-year history of the INL site, DOE and its predecessor agencies designed, built, and operated about 50 different conceptual reactors. All of these reactors have been prototype, low-power critical, or test reactors. INL operated no weapons production or commercial power reactors.

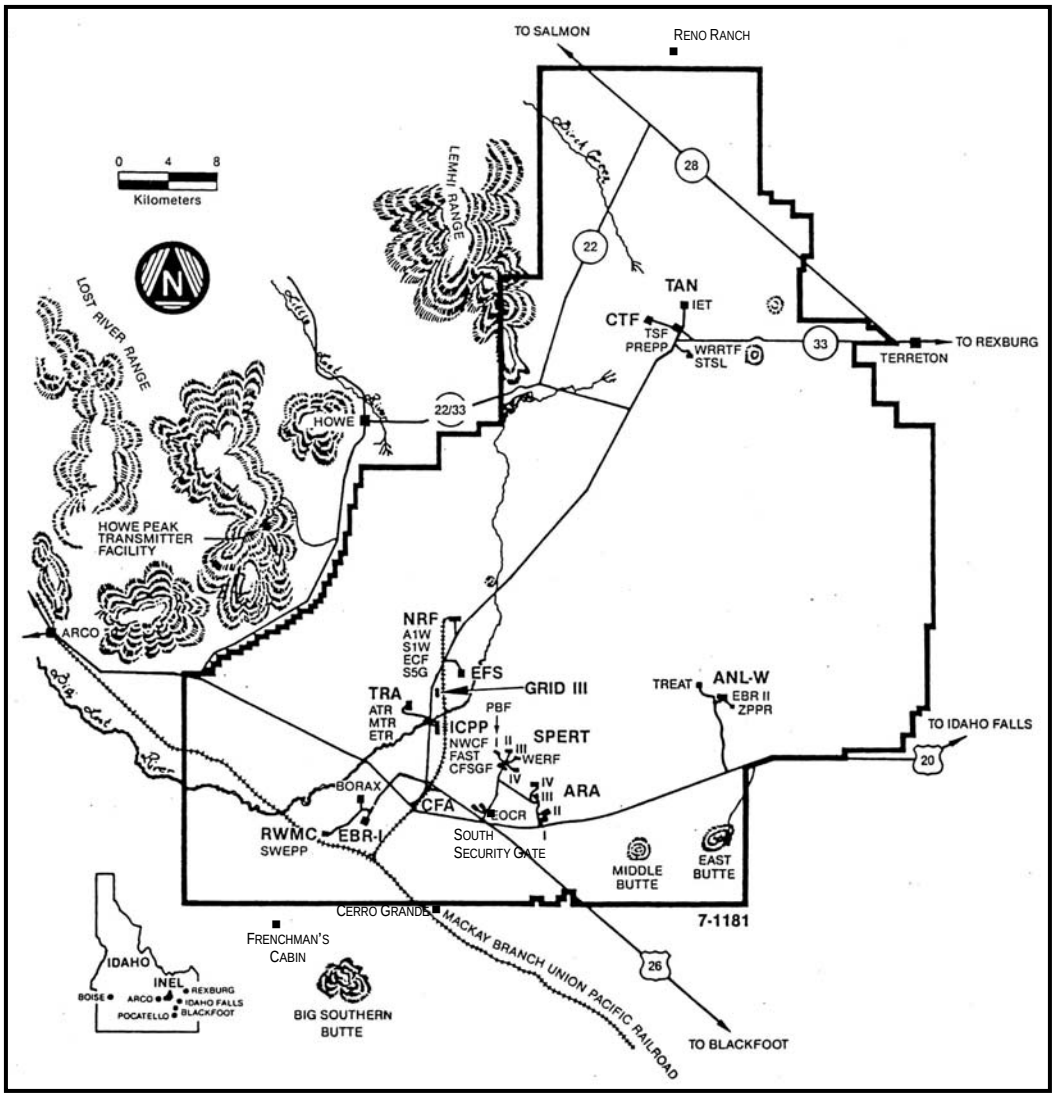

Figure 4-1. INL site map (Hoff, Chew, and Rope 1987, p.46).

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 INL site at Test Area North (TAN); (2) test reactors (Materials Test Reactor, Engineering Test Reactor, and Advanced Test Reactor), 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 (Osloond 1965; Osloond and Newcomb 1969).

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, which was constructed in the early 1950s, began processing nuclear fuel in February 1953 and continued until 1992 (Knecht et al. 1997, p 4). Throughout its history, the *Chem Plant*, or CPP, as it is commonly known, has reprocessed fuel from test reactors at INL, zirconium-clad fuel reclaimed from various reactors, stainless-steel-clad fuel from EBR-II, and fuel from many AEC test reactors 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 INL (DOE 1991;

Till et al. 2002). Through the years that Environmental Monitoring Reports (EMRs) have been published, INTEC airborne effluents have been attributed to creation of the majority of the INL boundary dose (Till et al. 2002). Therefore, it should be suspected that INTEC airborne effluent would also be responsible for the largest INL worker doses. Calculations performed for this TBD show that although ICPP airborne effluent is the most radiologically significant release at INL, the impact to all facility workers is significantly below the allowable and acceptable limits (Peterson 2004). Figure 4-1 shows INL facility locations, including the EBR-I and ANL-W facilities.

From the beginning of operations at the INL 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 mi^2, there was ample room to place facilities with this principle in mind. Because the site has an average elevation of 5,000 ft 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 INL and this meteorological characteristic have been satisfactorily effective in maintaining the operational isolation of each facility [1].

Beginning with the GE-ANP Program, which began in the early 1950s, the INL site has had the capability of plume tracking by aircraft. The local National Oceanic and Atmospheric Administration (NOAA) field office, which was dedicated to INL needs and requirements, provided plume projection capabilities 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, Fission Product Field Release Test, 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 [2].

INL 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 INL boundary from airborne releases that had occurred since the beginning of site operations (DOE 1991). With the help of NOAA, which had hourly meteorological data from 1956 to that time, INL 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 (RSAC-4) program (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, which was conducted by Radiological Assessment Corporation at the request of the Centers for Disease Control and Prevention and the State of Idaho, evaluated the methodology by which the RSAC-4 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” (Till et al. 2002, p. 57).

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

EBR-I was the first reactor to operate at the site. It and BORAX-I through -V, which were in the southwestern corner of INL, were operated for the AEC Chicago Operations Office by the University of Chicago as ANL-W. These low-power reactors produced essentially no radioactive airborne effluent (Osloond 1965; Osloond and Newcomb 1969). As the EBR-I and BORAX programs were completed, ANL-W relocated to the eastern section of the INL site where the locations of EBR-II, the TREAT Facility, the Zero Power Plutonium Reactor, etc., are indicated in Figure 4-1. 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 document apply to individuals stationed at the ANL-W facility – at the southwest location through 1965 and at the southeast location beginning in 1966 [3].

Attributions and annotations, indicated by bracketed callouts and used to identify the source, justification, or clarification of the associated information, are presented in Section 4.5.

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 INL during the history of the site (Till et al. 2002). For more discussion of these releases and their relationship to less significant releases, see Peterson (2004), DOE (1991), or Till et al. (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 type S and M materials being the predominant forms [4]. 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 a vapor intake of type F materials [5].

4.2.1 Operational Releases

Estimation of onsite concentrations of radionuclides and resulting potential intakes from operational releases at INL facility locations employs the same methodology used to determine offsite concentrations for site annual EMRs. The release for each year of operation is exactly the same as that documented in the 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 INL facility were picked from the annual average mesoscale dispersion isopleths of ground-level air concentrations in the EMRs, as described in the 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 INL upgraded the telemetry system. For years before 1973, the TBD analysis used a 9-year average of mesoscale dispersion isopleths of ground-level air concentrations (DOE 1991); see Figure 4-2. For 1993 to 2002, the analysis used annual average mesoscale isopleths from the annual environmental reports (Mitchell 1994, 1995, 1996, 1997; Evans et. al. 1998,; Saffel, et. al. 2000; Stoller 2002a,b,c, 2004, 2005) to calculate the facility annual concentration.

Of the many facilities on INL, 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 the present) have been

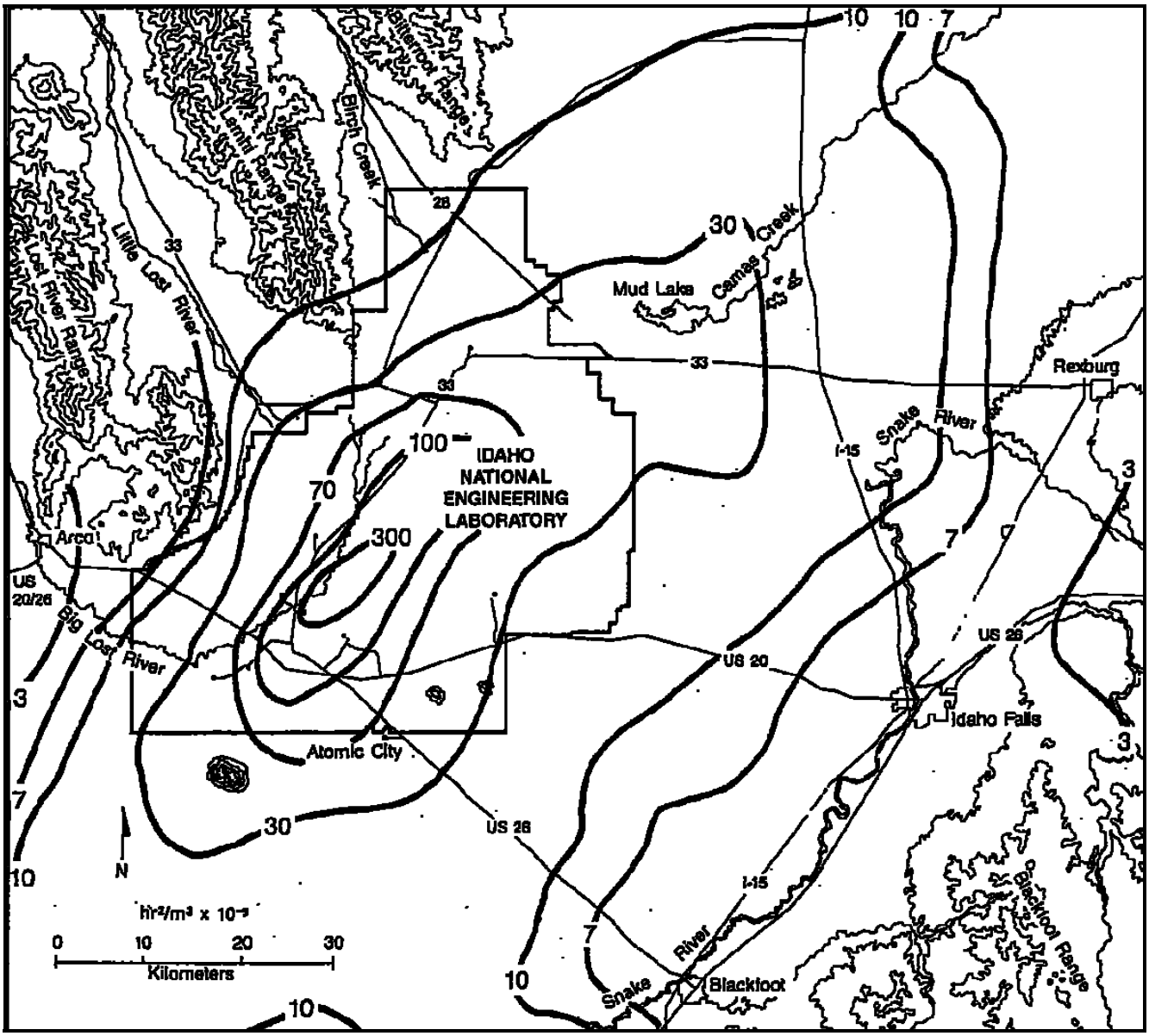

Figure 4-2. Nine-year (1974 to 1983 excluding 1978) average mesoscale dispersion isopleths of air concentration at ground level ($hr^2/m^3 \times 10^{-9}$) normalized to unit release rate (DOE 1991).

extracted from the annual EMRs and converted from the normalized annual concentration[2] (hr^2/m^3) to concentrations (becquerels per cubic meter) and multiplied by 2.4×10^3 m^3/yr (the amount of air breathed occupationally per year) to produce activity inhaled per year (becquerels) for an occupational individual. Table 4-1 lists these values.

The annual inhaled quantities (becquerels per year) in Table 4-1 are based on known and published INL annual airborne emissions. The following discussion provides information from NRTS/INEL/ INEEL/INL documentation on facility environmental sampling and monitoring and on data that can be

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

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 [7]. The early IDO Health and Safety (H&S) Division Annual Reports document

Table 4-1. Intake (Bq/yr) by year for ANL-W, 1952 to 2005 [6].

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
2003	9.4E-8	1.2E-2	2.4E-2	1.0E-5	0.0E+0	0.0E+0	0.0E+0	0.0E+0	0.0E+0
2004	1.8E-5	1.2E-4	4.9E-4	4.0E-5	5.6E-5	0.0E+0	5.9E-6	1.1E-3	0.0E+0

many studies for defining radionuclide concentrations in the vicinity of different facilities. These 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 H&S Division (AEC 1964) contains some facility environmental monitoring data. INL developed a routine facility environmental monitoring program between 1963 and 1970. In 1968 and 1969, formal EMRs (AEC 1968, 1969a,1969b, 1970a) reported alpha, beta, and ^{131}I concentrations that can be correlated with the data of Table 4-1. The 1970 EMR (AEC 1970b, 1971) 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 are available and usable for this correlation.

Table 4-2 lists results of the comparison. Because of the large variation in measurements, the ratio of values calculated from the EMRs (Table 4-2, column 5) to those derived from the releases in Table 4-1 is not well behaved. Three of the 13 values with uncertainties overlap. The geometric mean of the remainder of values is 0.36 with a geometric standard deviation of 5.3. Nevertheless, this comparison provides confidence in the results of Table 4-1. The value at the upper 84% confidence value 1.9 (0.36 × 5.3) is less than the default geometric standard deviation of 3 assumed for environmental results [8].

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

Year	Activity type	Average annual concentration	Reference for Col. 3	Annual inhaled quantity (Bq)	INL TBD table inhaled quantity (Bq)
1963	β-γ	17 pCi/m^3		1,510	1,310[a]
	Pu-239	0.6 fCi/m^3		0.05	0.014[a]
1968	α	2.2 fCi/m^3	AEC 1969a, p. 14	0.18	0.01[a]
	β	0.64 pCi/m^3	AEC 1969a, p. 14	56	337[a]
	I-131	<0.08 pCi/m^3	AEC 1969a, p. 14	<7.1	1.4[a]
1969	α	0.003 pCi/m^3	AEC 1970a, p. 14	0.27	2.4E-3[a]
	β	0.501 pCi/m^3	AEC 1970a, p. 14	44	118[a]
	I-131	<0.014 pCi/m^3	AEC 1970a, p. 14	<1.2	2.1[a]
1970	Gross β	0.6 pCi/m^3	AEC 1971, p. 5	53	74[a]
	Max. gr. β at CFA	0.81 pCi/m^3	AEC 1971, p. 5	72	74[a]
1973	Gross β	95 ±42 fCi/m^3	AEC 1974, p. 14	8.4 ±3.7	0.8[a]
EBR-I	Sr-90	3.4 ±3.0 fCi/m^3	AEC 1974, p. 16	0.3 ±0.27	0.15[b]
	Nb-95	1.0–2.5 fCi/m^3	AEC 1974, p. 16	0.09–0.22	---
	Cs-137	7–17 fCi/m^3	AEC 1974, p. 16	0.6–0.5	---
	Ce-144	4–8 fCi/m^3	AEC 1974, p. 16	0.36–0.71	0.057[b]
EFS	Sr-90	5.9 ±8.6 fCi/m^3	AEC 1974, p. 16	0.52 ±0.76	0.15[a]
	Nb-95	0.9–2.4 fCi/m^3	AEC 1974, p. 16	0.08–0.21	---
	Ru-106	6–9.8 fCi/m^3	AEC 1974, p. 16	0.53–0.87	0.27[a]
	Cs-134	0.8–1.6 fCi/m^3	AEC 1974, p. 16	0.07–0.14	---
	Cs-137	17–27 fCi/m^3	AEC 1974, p. 16	1.5–2.4	---
1976[c]	Gross β	30–60 fCi/m^3	ERDA 1977b	2.5– 5.3	0.6-25[d]
1986	Kr-85 at CFA	37 ±15 pCi/m^3	Hoff, Chew, and Rope 1987	3,290 ±1,330	890[e]
1988	Kr-85 at CFA	108 ±69 pCi/m^3	Hoff, Mitchell, and Moore 1989	9,770 ±6,130	14,000[e]
1990	Kr-85 at CFA	27.7 pCi/m^3	Hoff et al. 1991	2,400	690[e]

a. Values from the INEEL Environmental Occupational Dose TBD (ORAUT 2004, Table 4-3) for CFA.
b. Values from ORAUT (2004, Table 4-5) for the Radioactive Waste Management Complex (RWMC) because 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 Peterson (2004) that contained concentrations of all INL-released radionuclides.

Figure 4-3 shows the variation of INL environmental sampling results for 1978 through 1986, which is typical for earlier and for later years (Hoff, Chew, and Rope 1987). This figure also shows the close correlation of environmental sample results acquired at "distant communities" and those acquired at INL facilities and the effect of foreign nuclear tests and the Chernobyl reactor accident on INL environmental sampling results. As shown on this figure, the INL 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 EMRs show the same correlation for the years before 1978 and after 1986. Nearly all of the greater perturbations in facility and distant community concentrations correlate with fallout from nuclear tests [9]. There was no discernible evidence that facility effluent or resuspension affected facility concentrations.

4.2.2 Episodic Releases at INL

Of the 108 episodic releases (accidents and planned tests) analyzed in DOE (1991), only 16 had the potential to affect other INL facilities [10]. Only 9 of the 16 events could have affected the EBR-I facility [11]. Section 4.1.2.2 describes three of these events, two of the three criticalities at ICPP, and the Fuel Element Cutting Facility (FECF) Filter Break, which also occurred at ICPP. The other six 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 one or more onsite facility. For example, the FECF Filter Break occurred at ICPP and clearly contaminated an area south of ICPP (AEC 1960). 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 EBR-I [12].

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 [13]. The following sections discuss these events. The Stationary Low-Power Reactor No. 1 (SL-1) accident, which was widely publicized, is included only to show it did not affect any other facility at INL.

An effort has been made to reduce the number of radionuclides in the release analysis 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 [14]. When viewed together, the episodic events fall into three categories: criticalities that involved fresh fission products that have relatively short half-life in comparison to radionuclides released from the Fuel Element Burn Tests, for example; releases that involved long half-life, aged fission products (Fuel Element Burn Tests and the release from the FECF Filter Break); and releases from the remaining IETs that released short half-life radionuclides, which are generally characterized as fresh fission products, and long half-life radionuclides, which are characterized as aged fission products [15]. The latter category is unique to the GE-ANP Program because of the "direct-to-air" conversion nature of the tests [16]. 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 [17].

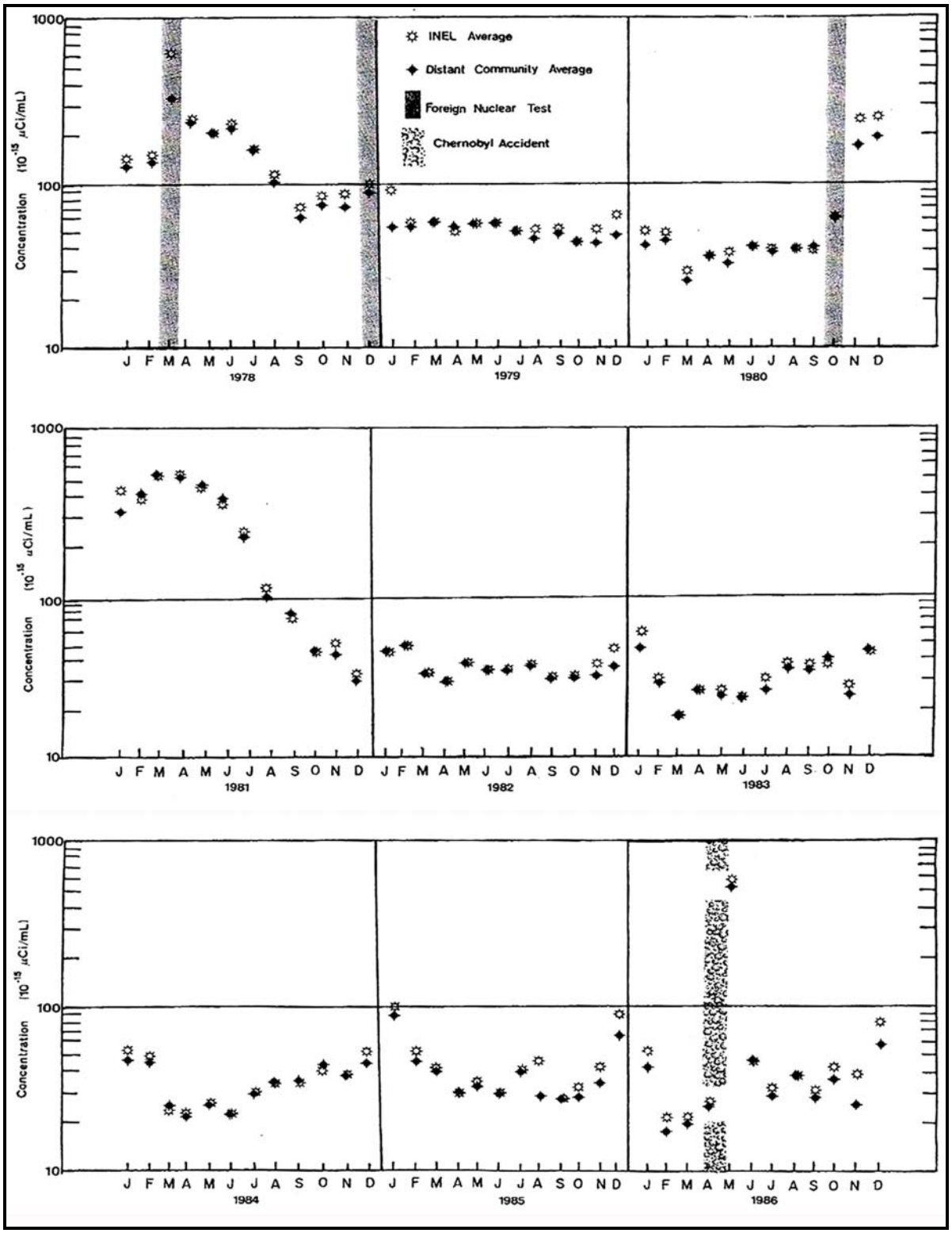

Figure 4-3. Onsite and distant particulate beta concentrations in air (Hoff, Chew, and Rope 1987, p. 17).

4.2.2.1 Stationary Low-Power Reactor No. 1 Accident

One significant accident at INL in the last 51 years released substantial amounts of RM to the environment. On January 3, 1961, a steam explosion at the SL-1 facility [near Auxiliary Reactor Area 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 INL facility with the effluent of RM. The effluent traveled to the south of the facility, as shown in Figure 4-4 (DOE 1991).

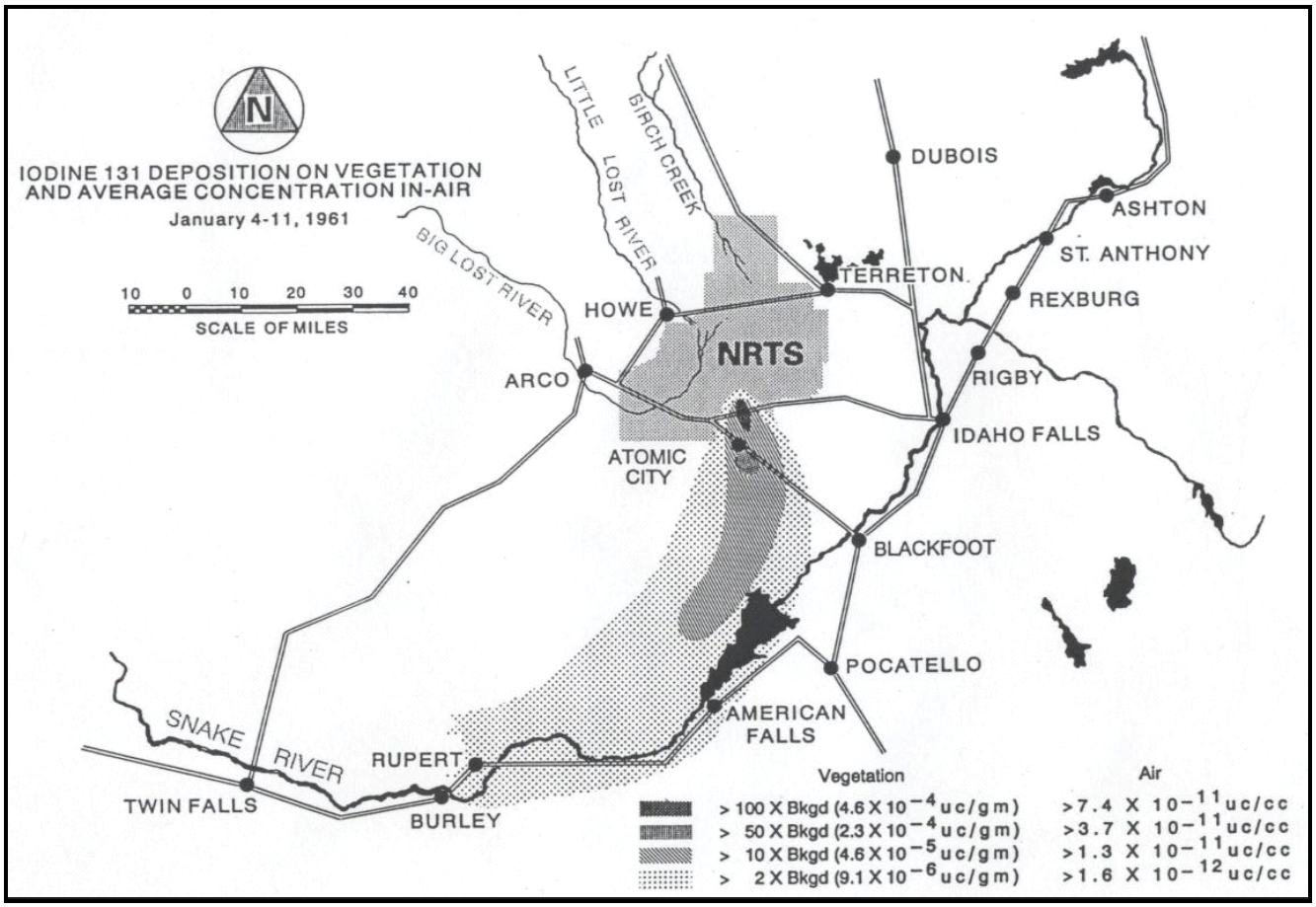

Figure 4-4. Dispersion coefficient contours for SL-1 accident (redrafted from Horan and Gammill 1961).

4.2.2.2 Criticality and Accident Occurrences at Idaho Chemical Processing Plant

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, as described in Sections 4.2.2.2.1, 4.2.2.2.2, and 4.2.2.2.3, respectively. The three criticalities released RM during or shortly after the event. In all cases, the effluent was transported to the south-southwest and potentially exposed personnel at the EBR-I facility because that facility was postulated to be in the

plume path. The 1978 criticality, which released essentially just noble fission gases that were produced during the criticality, occurred after ANL-W operations had moved to the present southeast site location and, thus, did not affect personnel at the new ANL-W location. The FECF Filter Break accident, which also occurred at the ICPP, has been postulated to have affected the EBR-I facility and to have been in the path of the release plume. Analyses that were favorable to claimants, described below, defined the amount of potential radiological exposure that could have occurred to an individual at this location.

4.2.2.2.1 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 calculated internal and measured external radiological doses for plant personnel who were involved in the incident.

For the calculation of intakes for this incident, meteorological conditions were modeled so the X/Q at 22 kilometers matched the value calculated for Frenchman's Cabin (south of INL, 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 [18].

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
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 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 on 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, INL conducted decontamination operations in the FECF. Acid fumes from the decontamination operations caused failure of the FECF exhaust filters, which resulted in the release of particulate activity to the south of ICPP (AEC 1959).

Approximately 100 Ci of long half-life particulate RM were released over approximately 200 acres (AEC 1959). The released RM 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 [19].

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 under the GE-ANP Program at TAN that potentially affected the EBR-I facility.

4.2.2.3.1 Initial Engine Test 14

IET 14 was the eighth nuclear test that was 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 that operated 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 that were 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. These intakes are for a total exposure period of 26 days.

Table 4-5. Intakes (Bq/event) at EBR-I for initial engine tests at INL [20].

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(B) 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 were 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 potential 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 fission product release for this test was divided into two periods based on a review of effluent monitoring data. The period from June 3 to 15, 1959, was considered to be an operation before the development of significant fuel sheet blisters. The second period was from June 16 to 24, 1959, when effects of blistering were clearly observed.

According to the meteorology of the testing period, the second period affected Frenchman's Cabin. Therefore, this analysis addressed the radiological impact on the EBR-I. Table 4-5 lists the release of fission products, which corresponds with the Part B operation release from 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 intakes are for the total 9-day exposure period.

4.2.2.3.3 Initial Engine Test 17(B)

IET 17(B) 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 that exceeded 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 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(A), conducted between February 9 and April 30, 1960, was a test series in the HTRE-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 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(A) was 2,892 Ci. The release for this test was modeled as was that 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 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(A), performed between November 15 and December 19, 1960, was an extension of the Phase II testing program in IET 18. The test was conducted in the HTRE-3 reactor configuration. Releases of airborne radioactivity that correspond 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 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 (A), conducted in HTRE-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 that exceeded 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 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 [21]. In general, direct beta/gamma radiation from the facility increases with time because the general contamination of the area increases. In addition, as a facility ages, radioactive sources tend to accumulate at the facility, which causes the general background to increase with time [22]. A responsible H&S organization observes and curbs such trends to prevent personnel exposures from increasing unnecessarily. The following sections discuss facility fenceline 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 Fenceline Annual Doses

Before 1970, many film badge or TLD measurements occurred inside the INL 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. At first, the badges were retrieved and read once a month. The frequency changed to 6 weeks in 1962 and then back to monthly in 1963 (AEC 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 (AEC 1968). 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 INL before operations began between 100 and 150 mrem/yr, the monthly value of 8 to 13 mrem is at the detection limit of the film badge or TLD (i.e., within the deviation of the 10-mrem limit). Therefore, the uncertainty for monthly exchanges is higher than for less frequent exchanges.

Facility and fence monitoring 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 fenceline TLD measurements, made on a 6-month basis with five TLDs at each facility position, are available in the quarterly Environmental Monitoring Data reports (Walker 1972, 1973a,b, 1974a,b, 1975a,b,c; Dahl 1975; Sill 1976; Bills 1976; Williamson 1976a,b, 1977a,b,c, 1978a,b; DOE 1978a,b,c, 1979a,b,c,d, 1980a,b,c,d, 1981a,b,c,d, 1982a,b,c,d, 1983a,b,c,d, 1984). Figure 4-5 (from Williamson 1976a) shows that uncertainty can vary from less than 10% to 20% for a given set of readings. At each of the 34 monitoring locations, there are 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 readings were assigned a 2-sigma uncertainty of 16% to 20%, and 18.4% of the readings were assigned a 2-sigma uncertainty of 11% to 15%. However, 80% of the 136 values ascribed to 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 (Williamson 1976a, p. 38).

To supply facility values for 1952 to 1972, the highest 6-month value from April 1972 to April 1973 for a facility was multiplied by 2 and applied to each year between 1952 and 1972. (Not all the listed facilities had commenced operation or even existed in 1952.) Facility fenceline TLD measurements could not be found for 1984 through 1992, but for 1993 and beyond such measurements were included in the EMRs (Mitchell 1994; Mitchell, Peterson, and Hoff 1995; Mitchell et al. 1996, 1997; Evans et al. 1998; Saffel et al. 2000; Stoller 2002a,b,c, 2003). For the period from 1984 to 1992 when TLD measurements were missing, reasonable interpolations were used to provide the missing values [23]. In addition, background TLD measurements that correspond with the facility fenceline TLD measuring periods were recorded in the EMRs. All reduced facility fenceline TLD data (facility fenceline data minus background) in the EMRs are listed in Table 4-6 (AEC 1971, 1972, 1973, 1974; ERDA 1975, 1976a, 1976b, 1977b; DOE 1978d, 1979e, 1980e, 1981e, 1982e, 1983e; Hoff, Chew, and Dickson 1984; Mitchell 1994; Mitchell, Peterson, and Hoff 1995; Mitchell et al. 1996, 1997; Evans et al. 1998; Saffel et al. 2000; Stoller 2002a,b,c, 2003, 2004, 2005). A more detailed discussion of the data is included in Peterson (2004).

4.3.2 Facility Air Immersion Doses

ANL-W 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 fenceline TLD doses in Table 4-6.

Table 4-6. ANL-W facility fenceline direct gamma values (TLD minus background) (mR) [25].

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
2003	20	11	117
2004	23	10	118

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

4.4 UNCERTAINTY

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

4.4.1 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.

4.4.2 Episodic Releases

As described in the INELHDE (DOE 1991), the episodic releases are a maximum reasonable value that is based on the amount of material available to be released and the conditions of the test. For such a release, the inhaled quantities (in becquerels) 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 that was considered for a particular episodic event might be low by as much as a factor of 3 [24].

4.4.3 Film Badge and Thermoluminescent Dosimeter 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) [26]. The data for 1965 to 1972 in Table 4-6 is based on the highest 6-month TLD values of 1972 for the respective facility. Although the GE-ANP IETs 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 [27]. However, after 1967 when facility fenceline measurements were routinely made with TLDs, with five TLDs at a given location the uncertainty is generally estimated at less than 10% [28]. Less than about 20% of the time, these measurements have an estimated level of uncertainty as high as 20% [29].

4.5 ATTRIBUTIONS AND ANNOTATIONS

Where appropriate in this document, bracketed callouts have been inserted to indicate information, conclusions, and recommendations provided to assist in the process of worker dose reconstruction. These callouts are listed here in the Attributions and Annotations section, with information to identify the source and justification for each associated item. Conventional References, which are provided in the next section of this document, link data, quotations, and other information to documents available for review on the Project's Site Research Database.

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

[1] Peterson, Henry. Principal Health Physicist. Intrepid Technology and Resources. September – December 2003.
Although major events at one facility have sometimes been measured at nearby facilities, they have never necessitated emergency measures.

[2] Peterson, Henry. Principal Health Physicist. Intrepid Technology and Resources. September – December 2003.
Tests were conducted over the GRID III when the meteorological conditions guaranteed the effluent would not affect other INL facilities. The monitoring grid was directed to the northeast, which was not toward any other facility at the INL site.

[3] Peterson, Henry. Principal Health Physicist. Intrepid Technology and Resources. September – December 2003.
Because the ANL-W facility and personnel were moved from the southwest corner to the southeast corner of the INL site in 1966, it is good sense to determine the effluents that would be directed by meteorological diffusion to affect the facility in a new location.

[4] Peterson, Henry. Principal Health Physicist. Intrepid Technology and Resources. September – December 2003.
This is a recommendation to use the absorption types that are most predominant. This recommendation is supported by the reference Peterson 2004.

[5] Peterson, Henry, Principal Health Physicist. Intrepid Technology and Resources. September – December 2003.
This recommendation to use a vapor intake of absorption type F material is based on the default value in ICRP Publication 66 (ICRP 1994).

[6] Peterson, Henry, Principal Health Physicist. Intrepid Technology and Resources. September – December 2003.
This limited list of radionuclides provides more than 95% of the effective dose equivalent based on ICRP Publication 68 (ICRP 1995). Every radionuclide in the list of radionuclides from Peterson (2004) was multiplied by the effective dose coefficient for inhalation of 5-μm activity median aerodynamic diameter (AMAD) from Annex B of ICRP Publication 68 and then added together. In steps, nuclides were eliminated based on effective dose being less than 0.1% of the total, then 0.3%, and then 0.4%. The fraction of effective dose equivalent lost is less than 5%, but the activity lost is considerably more.

[7] Peterson, Henry, Principal Health Physicist. Intrepid Technology and Resources. September – December 2003.
An example of such analysis is Ebersole (1956).

[8] Peterson, Henry, Principal Health Physicist. Intrepid Technology and Resources. September – December 2003.
The default geometric standard deviation of 3 has been confirmed by email, C. Bloom to N. Rohrig of 11/09/2006.

[9] Peterson, Henry, Principal Health Physicist. Intrepid Technology and Resources. September – December 2003.
This statement is a conclusion from a review of the environmental data for onsite and distant offsite locations.

[10] Peterson, Henry, Principal Health Physicist. Intrepid Technology and Resources. September – December 2003.
When the analysis of DOE (1991) was completed, the meteorological diffusion trajectories were reviewed to determine which INL facilities were affected. The result of these trajectories was that only 16 releases had the potential to affect other INL facilities.

[11] Peterson, Henry, Principal Health Physicist. Intrepid Technology and Resources. September – December 2003.
Through a review, conducted during the analysis described in DOE (1991), of the meteorological trajectories of all of the radiological releases, only nine of the releases affected the ANL-W facility – whether the ANL-W facility was to the southwest or at the southeast location, depending on the date of the release and location of the ANL-W facility.

[12] Peterson, Henry, Principal Health Physicist. Intrepid Technology and Resources. September – December 2003.
The radiological impact analysis was performed to determine the diffusion at the EBR-I facility.

[13] Peterson, Henry, Principal Health Physicist. Intrepid Technology and Resources. September – December 2003.
The radiological impact analysis was performed to determine the diffusion at the EBR-I facility.

[14] Peterson, Henry, Principal Health Physicist. Intrepid Technology and Resources. September – December 2003.
“Fresh” fission products are those that result from a recent fission process; “aged” fission products are those that remain after a period of decay, which is measured in terms of months or years. A criticality accident that occurs in a solution of uranium fuel, such as those criticalities that occurred at the ICPP, involves aged/decayed fuel and, thus, aged or decayed fission products. However, the driving mechanisms for release are the heat and energy created by the criticality event, which create fresh fission products such as the short half-life, volatile noble gases and iodines that make up the majority of the fission product release from such an event.

[17] Peterson, Henry, Principal Health Physicist. Intrepid Technology and Resources. September – December 2003.
The “episodic” releases ranged from criticality accidents that occurred in aqueous solutions of aged fuel to tests that involved burning fuel elements that had been allowed to decay for months, which had essentially no fresh fission products remaining in them.

[16] Peterson, Henry, Principal Health Physicist. Intrepid Technology and Resources. September – December 2003.
The GE-ANP Program conducted tests in the HTRE cores that released fresh and aged fission products simply because the fuel cladding was destroyed and the fuel and the contained fission products were released directly to the atmosphere by way of the IET stack.

[17] Peterson, Henry, Principal Health Physicist. Intrepid Technology and Resources. September – December 2003.
In the case of each episodic event, the release to the environment was a result of the actual operational history of the fuel involved being programmed into the RSAC computer program so that the actual or best-estimate fuel fission product inventory was used in the calculation of the release to the environment. In the analysis for DOE (1991), the actual or best-estimate release might have involved as many as 50 or 60 radionuclides, but the great majority of the

resulting dose to a receptor was due to only a handful of radionuclides. The number of radionuclides was reduced as described in Peterson (2004) such that 95% of the original dose was preserved.

[18] Peterson, Henry, Principal Health Physicist. Intrepid Technology and Resources. September – December 2003.
This limited list of radionuclides provides more than 95% of the effective dose equivalent based on ICRP Publication 68 (ICRP 1995). Every radionuclide in the list of radionuclides from Peterson (2004) was multiplied by the effective dose coefficient for inhalation of 5-µm AMAD from Annex B of ICRP Publication 68 and then added together. In steps, nuclides were eliminated based on effective dose being less than 0.1% of the total, then 0.3%, and then 0.4%. The fraction of effective dose equivalent lost is less than 5%, but the activity lost is considerably more.

[19] Peterson, Henry, Principal Health Physicist. Intrepid Technology and Resources. September – December 2003.
This limited list of radionuclides provides more than 95% of the effective dose equivalent based on ICRP Publication 68 (ICRP 1995). Every radionuclide in the list of radionuclides from Peterson (2004) was multiplied by the effective dose coefficient for inhalation of 5-µm AMAD from Annex B of ICRP Publication 68 and then added together. In steps, nuclides were eliminated based on effective dose being less than 0.1% of the total, then 0.3%, and then 0.4%. The fraction of effective dose equivalent lost is less than 5%, but the activity lost is considerably more.

[20] Peterson, Henry, Principal Health Physicist. Intrepid Technology and Resources. September – December 2003.
This limited list of radionuclides provides more than 95% of the effective dose equivalent based on ICRP Publication 68 (ICRP 1995). Every radionuclide in the list of radionuclides from Peterson (2004) was multiplied by the effective dose coefficient for inhalation of 5-µm AMAD from Annex B of ICRP Publication 68 and then added together. In steps, nuclides were eliminated based on effective dose being less than 0.1% of the total, then 0.3%, and then 0.4%. The fraction of effective dose equivalent lost is less than 5%, but the activity lost is considerably more.

[21] Peterson, Henry, Principal Health Physicist. Intrepid Technology and Resources. September – December 2003.
The facilities at INL are spaced at distances such that direct radiation from one facility does not affect another facility. Therefore, the only other method for creation of a direct gamma or beta-gamma dose to personnel at the facility is for the facility to become engulfed in a cloud of RM, either from that facility or from another facility.

[22] Peterson, Henry, Principal Health Physicist. Intrepid Technology and Resources. September – December 2003.
As a facility ages, dirt and waste, including RM, generally accumulates in the facility. A good example of such accumulation is at a foundry or machine shop. If management follows best practices, these materials and wastes are removed as they accumulate; if not, the general background radiation, as well as clutter in the facility increases with time.

[23] Peterson, Henry, Principal Health Physicist. Intrepid Technology and Resources. September – December 2003.
This interpolation is described in Peterson (2004). For all facilities except TRA and ICPP, interpolated data is the average of TLD data for that facility from 1993 through 1999, zeros excluded. For TRA and ICPP, there is a downward trend in the readings with time. Therefore, for these two facilities, the bottom four values (1989 through 1992) are the averages of the seven values below, and the top five values (1984 through 1988) are the averages of the seven numbers directly above.

[24] Peterson, Henry, Principal Health Physicist. Intrepid Technology and Resources. September – December 2003.
A factor of 3 was chosen by the authors of the DOE (1991) study as the upper bound value by which the GE-ANP releases could be inaccurate.

[25] Peterson, Henry, Principal Health Physicist. Intrepid Technology and Resources. September – December 2003.
See the reference list.

[26] Peterson, Henry, Principal Health Physicist. Intrepid Technology and Resources. September – December 2003..
Environmental film badges were exchanged monthly or more often in the early days. With normal background about 10 mrem/mo and a minimum reporting level of 10 mrem, only large excursions would be measurable. TLDs that were exchanged semiannually changed that situation appreciably.

[27] Peterson, Henry, Principal Health Physicist. Intrepid Technology and Resources. September – December 2003.
There are multiple references to a test being authorized based on the prevailing and expected wind conditions. NOAA has been stationed in Idaho Falls to provide weather monitoring and test control.

[28] Peterson, Henry, Principal Health Physicist. Intrepid Technology and Resources. September – December 2003.
This is based on Figure 4-5, which is a typical set of results.

[29] Peterson, Henry, Principal Health Physicist. Intrepid Technology and Resources. September – December 2003.
This is based on Figure 4-5, which is a typical set of results.

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