TABLE OF CONTENTS

LIST OF TABLES

LIST OF FIGURES

RECORD OF ISSUE/REVISIONS

ISSUE AUTHORIZATION DATE	EFFECTIVE DATE	REV. NO.	DESCRIPTION
Draft	9/26/2003	00-A	New Technical Basis Document for the Idaho National Engineering and Environmental Laboratory – Occupationally Environmental Dose. Initiated by Norman Rohrig.
Draft	10/13/2003	00-B	Incorporates comments from Noel Savignac and Cindy Bloom. Initiated by Norman Rohrig.
Draft	12/05/2003	00-C	Incorporates comments from NIOSH, Craig Little, and Laura McDowell-Boyer. Initiated by Norman Rohrig.
Draft	1/16/2004	00-D	Incorporates comments from Cindy Bloom. Initiated by Norman Rohrig.
Draft	2/16/2004	00-E	Incorporates comments from NIOSH for concurrence. Initiated by Norman Rohrig.
03/30/2004	03/30/2004	00	First approved issue. Initiated by Norman Rohrig.

ACRONYMS AND ABBREVIATIONS

AEC	U.S. Atomic Energy Commission
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
Ci	curie
CPP	(Idaho) Chemical Processing Plant
CSC	Computer Science Center (a facility in Idaho Falls)
CTF	Core Test Facility
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
EOCR	Experimental Organic Cooled Reactor
ERDA	Energy Research and Development Administration
ESRF	Environmental Science and Research Foundation
ETR	Engineering Test Reactor
F	Fast (solubility rate)
FEBT	Fuel Element Burn Test
FECF	Fuel Element Cutting Facility
FP	Fission Product
FPFRT	Fission Product Field Release Test
GCRE	Gas-Cooled Reactor Experiment
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
IRC	INEL Research Center (a facility in Idaho Falls)

kW	killoWatts
LOFT	Loss of Fluid Test (Facility)
LPT	Low-Power Test (Facility)
LPTF	Low Power Test Facility
M	Moderate (solubility rate)
ML-1	Mobile Low-power Reactor No. 1
mR	milli Roentgen
mrem	millirem
MTR	Materials Testing Reactor
MW	megawatt
NCRP	National Council on Radiation Protection and Measurements
NOAA	National Oceanic and Atmospheric Administration
NRF	Naval Reactor Facility
NRTS	National Reactor Testing Station
OMRE	Organic Moderated Reactor Experiment
PBF	Power Burst Facility
PREPP	Process Experimental Pilot Plant
RAC	Radiological Assessment Corporation
RaLa	radioactive lanthanum
RM	radioactive material
RSAC	Radiological Safety Analysis Computer (program)
RWMC	Radioactive Waste Management Complex
RWMIS	Radioactive Waste Management Information Service
S	Slow (solubility rate)
SL-1	Stationary Low-Power Reactor No. 1
SMC	Specific Manufacturing Capability
STPF	Shield Test Pool Facility
SPERT	Special Power Excursion Reactor Test
TAN	Test Area North
TBD	Technical Basis Document
TLD	Thermoluminescent Dosimeter
TRA	Test Reactor Area
TREAT	Transient Reactor Test
TSF	Technical Support Facility
WERF	Waste Experimental Reduction Facility
WRRTF	Water Reactor Research Test Facility
ZPPR	Zero Power Plutonium Reactor

4.1 INTRODUCTION

This technical basis document (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 another facility. The releases discussed here have been divided into two components: (1) normal operational releases, and (2) episodic releases that generally are of short duration. These RM releases potentially represent unrecorded or missed doses, either as direct gamma or beta-gamma from immersion in the radioactive gaseous cloud, for those individuals who did not have personal dosimetry to record the dose, or as internal doses from RM inhalation.

This TBD also 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 the facility. INEEL facility fence-line doses (minus background) are presented for 11 locations.

As outlined and discussed in Part 2 of this INEEL Site Profile, the INEEL Site was chosen by the U.S. Atomic Energy Commission (AEC) as an isolated location for testing various reactor concepts. The Site is isolated from the public in two important aspects: (1) it is remotely located from population centers, and (2) it is hydrologically isolated because no surface streams originate on and flow to an offsite location and no streams cross the Site. Although the Site 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 measured at 5 to 15 feet per day.

During the 50-year history of the site about 50 different reactor concepts have been designed, built, and operated at INEEL. All of these reactors have been prototype, low-power critical, or test reactors; no weapons production or commercial power reactors have been operated at INEEL. 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 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) the Experimental Breeder Reactor II (EBR-II), at Argonne National Laboratory-West (ANL-W) at the south-eastern corner of the Site.

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 will be accounted for and discussed below, ICPP airborne releases have historically been the most radiologically significant releases at INEEL. Through the years that INEEL environmental monitoring reports have been published, ICPP airborne effluents have been attributed to creating the maximum INEEL boundary dose. Considering this fact, it should be suspected that ICPP airborne effluent would also be 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 workers is significantly below the allowable and acceptable limit. INEEL facility locations mentioned above are shown in Figure 4-1.

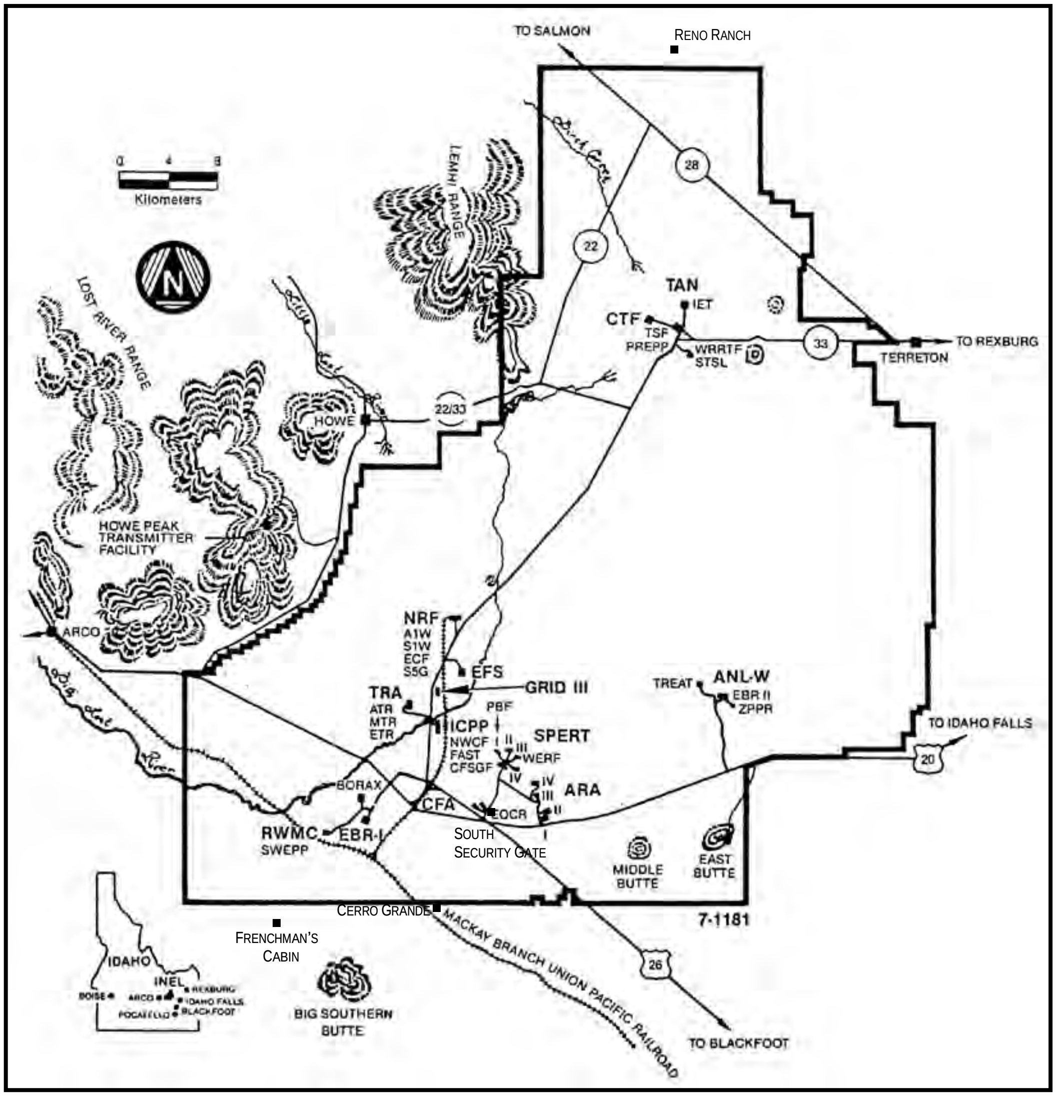

Figure 4-1. INEEL Site Map.

From the beginning of operations at the INEEL Site, facility locations were chosen 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 INEEL Site has an average elevation of 5,000 feet and is generally meteorologically characterized with 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 satisfactorily effective in maintaining the operational isolation of each facility.

Beginning with the GE-ANP program, which began in the early 1950s, the site has had the capability of plume tracking by aircraft. The local National Oceanic and Atmospheric Administration (NOAA) field office, which was dedicated to Site 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 [Controlled Environmental Release Test (CERT), Fission Product Field Release Test (FPFRT), Fuel Element Burn Tests A and B, etc.] were directed over an instrumented monitoring grid (GRID III), remotely located from other facilities, such that other onsite facilities were not affected by the release. In general, these tests were performed in support of a specific program (i.e., the FPFRT and the Fuel Element Burn Tests A and B were conducted to support the GE-ANP Program).

All of the airborne releases at INEEL that have occurred since the beginning of the Site were reviewed and analyzed as a result of a request from the U.S. Department of Energy (DOE). This request by the DOE Idaho Operations Office (IDO) was to evaluate the radiological impact to INEEL boundary individuals from airborne releases that had occurred since the beginning of operations at the Site. With the help of NOAA, which had hourly meteorological data from 1956 to that time, analyses were completed for all airborne releases that occurred at INEEL. The 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 basis for the releases considered in this TBD. In addition, all the 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 the Radiological Assessment Corporation (RAC 2002) at the request of the Centers for Disease Control and Prevention (CDC) and the State of Idaho, also evaluated the methodology by which the RSAC-4 computer program performs dose calculations against the National Council on Radiological Protection and Measurements (NCRP) methodology. 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 the current report to provide onsite concentrations due to episodic releases as well as other evaluations. For more information on the RSAC code, see Peterson (2004).

Figure 4-2 shows the chronology of NRTS/INEL/INEEL facilities and programs. A few comments on the development of facilities at INEEL will clarify some of the questions that arise when viewing the INEEL map in Figure 4-1 and putting it into context with the timeline in Figure 4-2.

The Experimental Breeder Reactor No. 1 (EBR-I) was the first reactor to operate at the Site. It and the Boiling Water Reactor Experiments (BORAX-I through –V) were in the southwestern corner of INEEL, operated under the AEC Chicago Operations Office by the University of Chicago as Argonne National Laboratory- West. These essentially low-power reactors produced very little radioactive airborne effluent. As the EBR-I and BORAX programs were completed, ANL-W relocated to the eastern section of INEEL where EBR-II, the Transient Reactor Test (TREAT) Facility, the Zero Power Plutonium Reactor (ZPPR), etc., are presently located. The EBR-I location is now a historic landmark. The Stationary Low-Power Reactor (SL-1), the Mobile Low-Power Reactor (ML-1), and the Gas-Cooled Reactor Experiment (GCRE), which were all operated for the U. S. Army, were at the Auxiliary Reactor Area (ARA). The NRTS Burial Ground became the Radioactive Waste Management Complex (RWMC). The short-lived Organic Moderated Reactor Experiment (OMRE) and Experimental Organic Cooled Reactor (EOCR) were at the EOCR location.

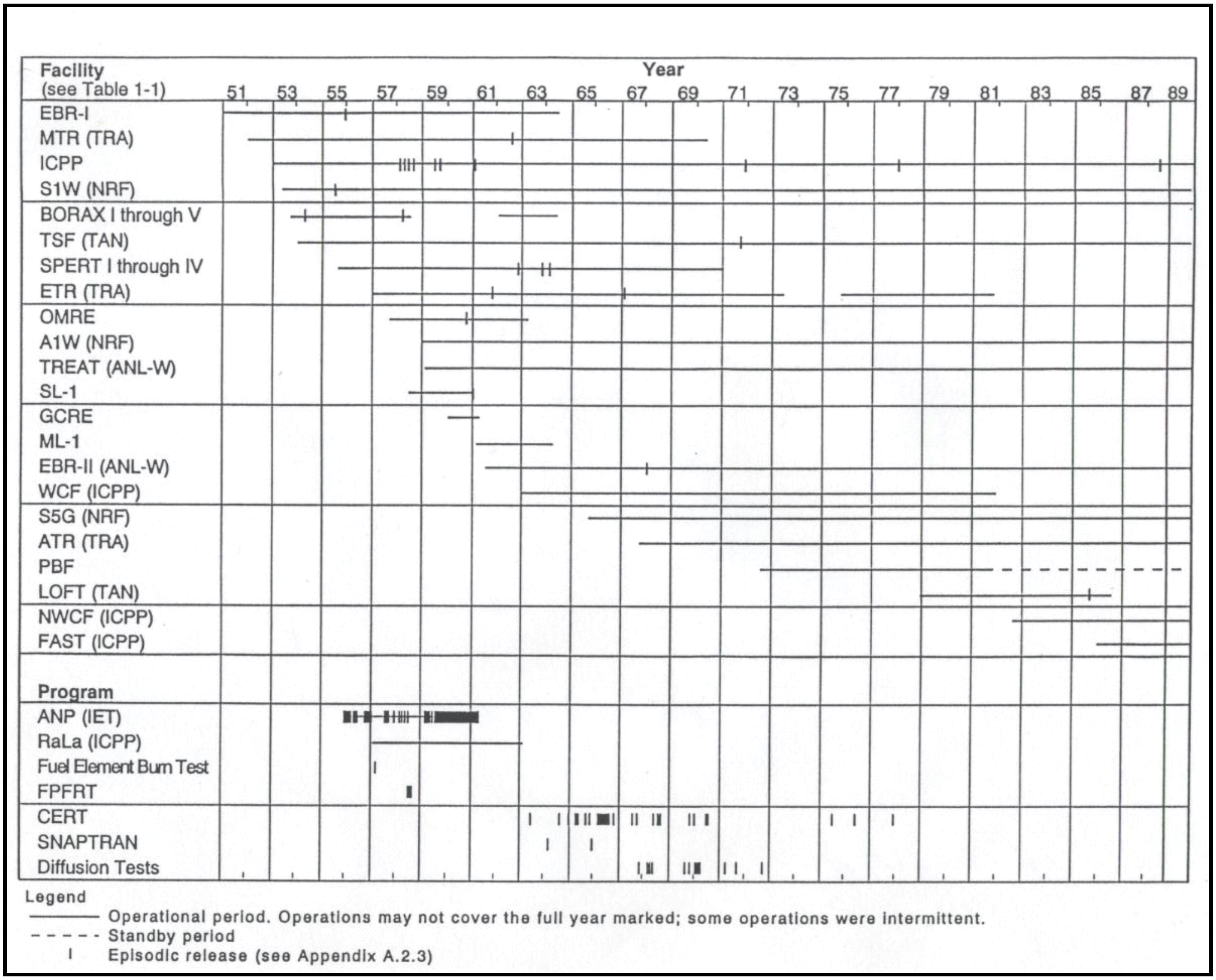

Figure 4-2. Chronology of NRTS/INEL/INEEL Facilities and Programs.

Essentially all of the major facility areas, the TRA, ICPP, TAN, Central Facilities Area (CFA), Naval Reactor Facility (NRF), Special Power Excursion Reactor Test (SPERT), and RWMC have operated since the early days of the Site. The major changes in these facility areas are the extent of operations at these facilities.

All inhaled quantities and concentrations referred to in this document apply to individuals stationed at the INEEL Site. DOE and INEEL contractor employees located principally in Idaho Falls, Idaho, in DOE/INEEL contractor facilities [Willow Creek Bldg., the INEL Research Center (IRC), the Computer Science Center (CSC), etc.] are not affected by Site activities and, thus, not subject to the inhaled quantities and concentrations.

4.2 INTERNAL INTAKES FROM ONSITE AIRBORNE RADIONUCLIDE CONCENTRATION

This section addresses onsite concentrations of radionuclides from normal operational releases and from shorter term releases such as those from criticality incidents that happened at ICPP and HTRE No. 3 during Initial Engine Test (IET) No. 13, Fuel Element Burn Tests, etc. 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), INELHDE (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, they should be considered "F" type materials.

4.2.1 Operational Releases

To determine onsite concentrations of radionuclides from operational releases at INEEL facility locations, the same methodology used to determine offsite concentrations for annual environmental monitoring reports is used. The release for each year of operation is exactly the same as that documented in INELHDE (DOE 1991) with one exception: an analysis was performed to reduce the number of radionuclides and yet retain those that contributed about 95% of the inhalation dose. This analysis, included in 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 the environmental monitoring reports published for INEEL as described in INELHDE (DOE 1991, Appendix B). As described in that Appendix, dispersion isopleths are available for the years beginning in 1973, with the exception of 1978, when the telemetry system was being upgraded. For the years prior to 1973, a 9-year average of mesoscale dispersion isopleths of ground-level air concentrations (Bowman 1984), shown in Figure 4-3, was used[1]. For 1993 to 2002, annual average mesoscale isopleths from the annual environmental reports (ESRF 1994, 1995, 1996, 1997, 1998, 2000; Stoller 2002a,b,c) were used to calculate the facility annual concentration.

Of the many facilities on INEEL, eight facility areas have been chosen for analysis: TAN, ICPP, TRA, RWMC, CFA, SPERT, ARA, and ANL-W. TAN includes IET, CTF, TSF, LOFT, SMC, WRRTF, PREPP, and LPTF, depending on the year of operation. The SPERT area includes WERF and PBF depending on the year of operation. Facilities such as Grid III and the Experimental Field Station (EFS) are inhabited infrequently and have not been included. These facilities were staffed with personnel who were normally employed at CFA. When an isopleth for a given year is chosen for a particular facility, such as SPERT, that isopleth is assumed to apply to PBF, SPERT-I, SPERT-II, etc. If a facility was between two isopleths, the higher-valued isopleth was chosen. Yearly isopleth values for each of the eight facilities 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) and multiplied by 2.4E+3 m^3/yr (the amount of air breathed occupationally per year) to produce activity inhaled per year (Bq) for an occupational individual. These are listed in Tables 4-1 through 4-8 for each of the facility areas.

The annual inhaled quantities (Bq/yr) provided in Tables 4-1 through 4-8 for each of 8 facility areas are based on known and published INEEL annual airborne emissions. The following discussion provides information that could be located in NRTS/INEL/INEEL documentation concerning facility environmental sampling/monitoring and provides data that can be compared with these calculations.

---

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

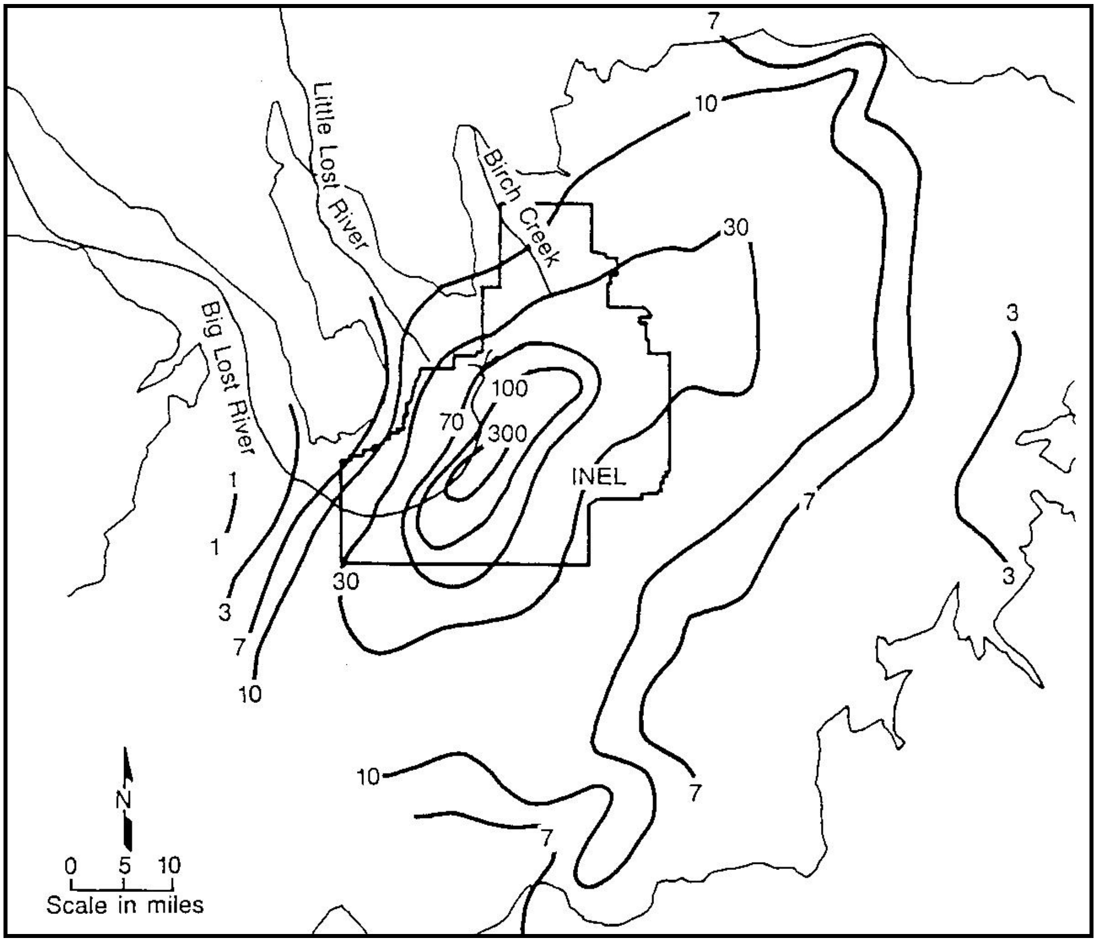

Figure 4-3. Nine-year (1974 - 1983) average mesoscale dispersion isopleths of air concentration at ground level ($hr^2/m^3 \times 10^{-9}$), normalized to unit release rate (Bowman 1984).

“Environmental” air sampling at the facility areas has been performed at least since the mid 1950s and was accomplished where airborne effluents were known or suspected to exist. The early IDO H&S Division Annual Reports document 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 not performed in a set facility location, or for a standard time duration. Some facility data is presented in the 1963 Annual Progress Report of the IDO Health and Safety Division and a routine facility environmental monitoring program is developed between 1963 and 1970. In 1968 and 1969, formal Environmental Monitoring Reports (EMRs) report alpha, beta, and I-131 concentrations that can be correlated with Table 4-1 through 4-8 values. The 1970 EMR reports gross beta values measured at CFA that can be correlated with Table 4-3 (CFA) values. EMRs between 1970 and 1990 were reviewed for data that could be used for this correlation. Results of the comparison, showing that the calculated values of Tables 4-1 through 4-8 are in reasonable agreement with measured values reported by EMRs, are provided in Table 4-9.

Figure 4-4 illustrates the variation of the INEEL Environmental Monitoring sampling results for the 9-year period 1978 through 1986. This figure also illustrates the close correlation of environmental sample results acquired at “distant communities” and those acquired at the INEEL facilities and the effect of foreign nuclear tests and the Chernobyl reactor accident on INEEL environmental sampling results. As shown on this figure for the 9-year period from 1978 through 1986, the INEEL average concentration has not differed from “distant community” concentrations by more than a factor of 2 for this 9-year period. Inspection of subsequent year’s Environmental Monitoring Reports shows the same is true for the years beyond 1986.

It is also interesting to note that the greater perturbations in the facility, and distant community, concentrations are nearly all correlated with “fallout” from nuclear tests. An example of such a perturbation, attributed to a September 26, 1976, atmospheric test conducted by the People’s Republic of China, is illustrated in the graph of Figure 4-5 (ERDA 1976) where the normal concentration was increased by a factor of 20 and had a 3-month influence on the average concentration for all of the air concentrations measured, on-site and off-site. In the history of air-monitoring at the INEEL, an operational release has never approached the magnitude of perturbation created by fallout as illustrated in the graph of Figure 4-5.

4.2.2 Episodic Releases at INEEL

4.2.2.1 SL-1 Reactor Accident

One significant accident at INEEL in the last 51 years released substantial amounts of radioactive material to the environment. The steam explosion at the SL-1 facility (near the present location of ARA II in Figure 4-1) killed three SL-1 personnel on January 3, 1961, and ruptured the SL-1 reactor vessel. This, in turn, propelled radioactive material 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 was 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 radioactive material. The effluent traveled to the south of the facility, as shown in Figure 4-6 (DOE 1991).

4.2.2.2 Criticalities

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 released essentially just the noble fission gases produced during the criticality; halogens and solids were removed by the ICPP exhaust filtering system (Casto 1980). However, the two earlier criticalities did release radioactive material during or shortly after the event; in both cases, the effluent was transported to the south-southwest and potentially exposed personnel at the southern end of ICPP and at CFA. A conservative analysis, described below, defined the amount of potential radiological exposure that may have occurred to an individual at these locations.

4.2.2.2.1 ICPP Criticality of October 16, 1959

On October 16, 1959, at approximately 3:00 a.m., the first 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) gives 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 that the X/Q at 22 km matched the value calculated for the Frenchman’s Cabin (located south of the INEEL as

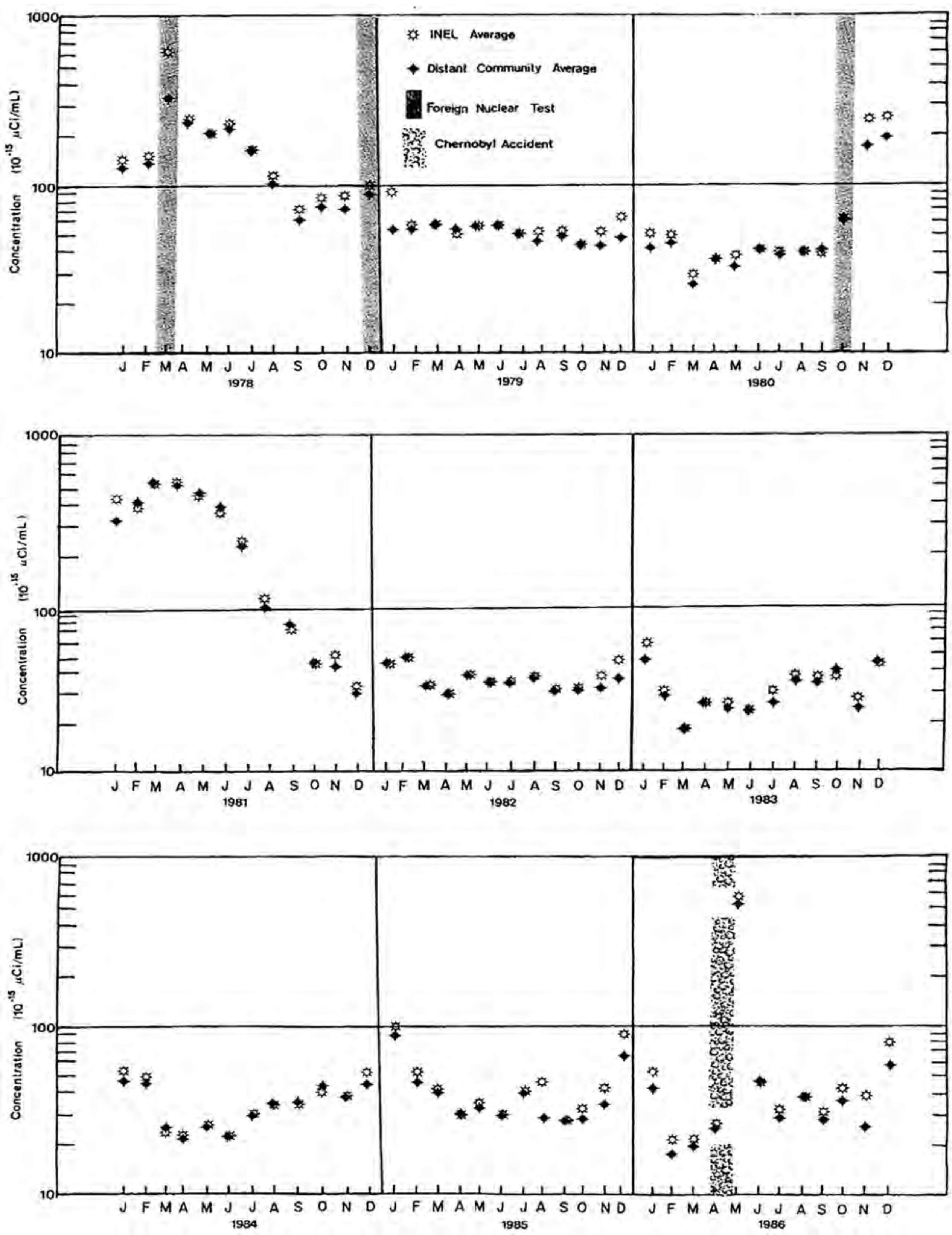

Figure 4-4. Onsite and Distant Particulate Beta Concentrations in Air

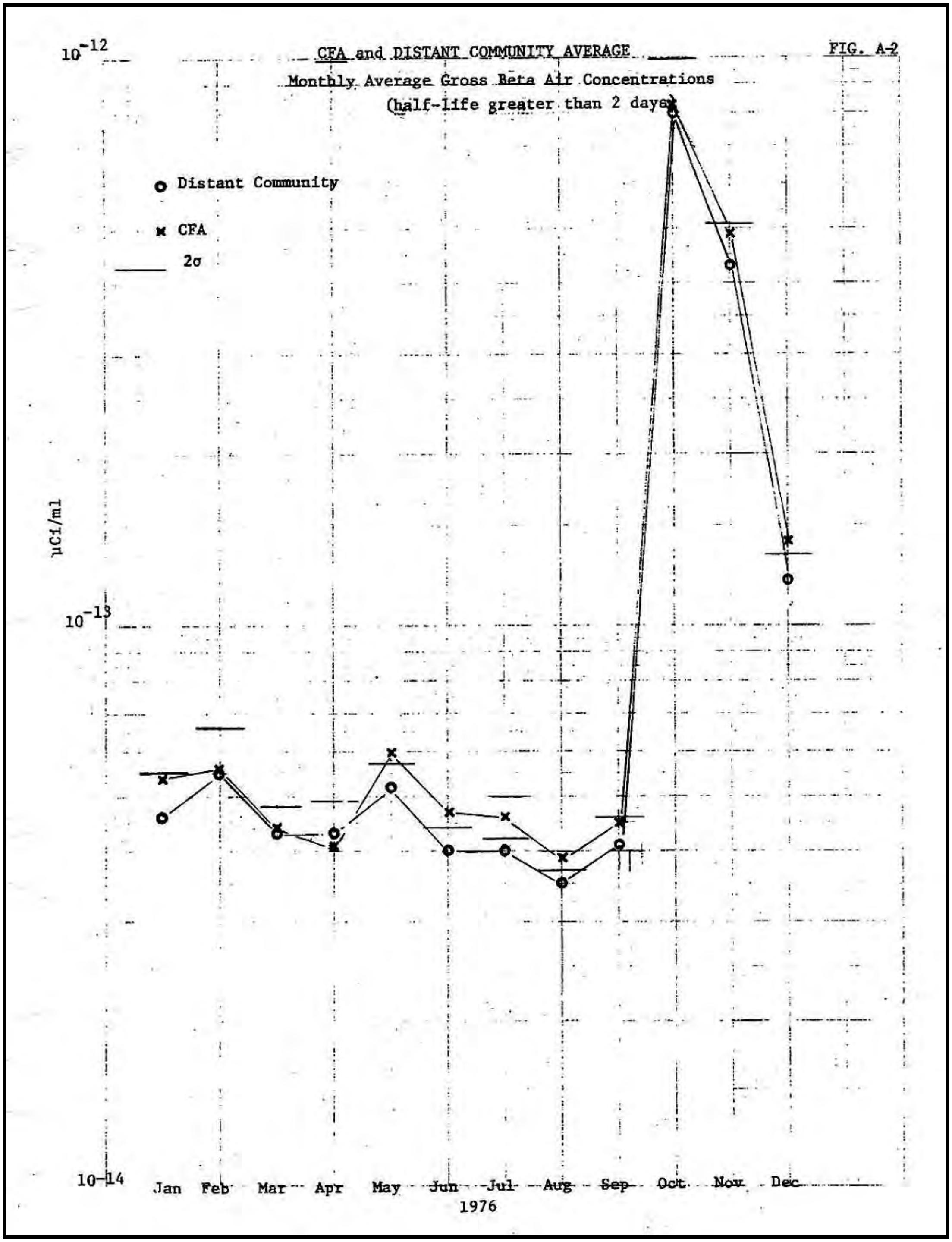

Figure 4-5: CFA Air Monitoring (Gross Beta) Data for 1976.

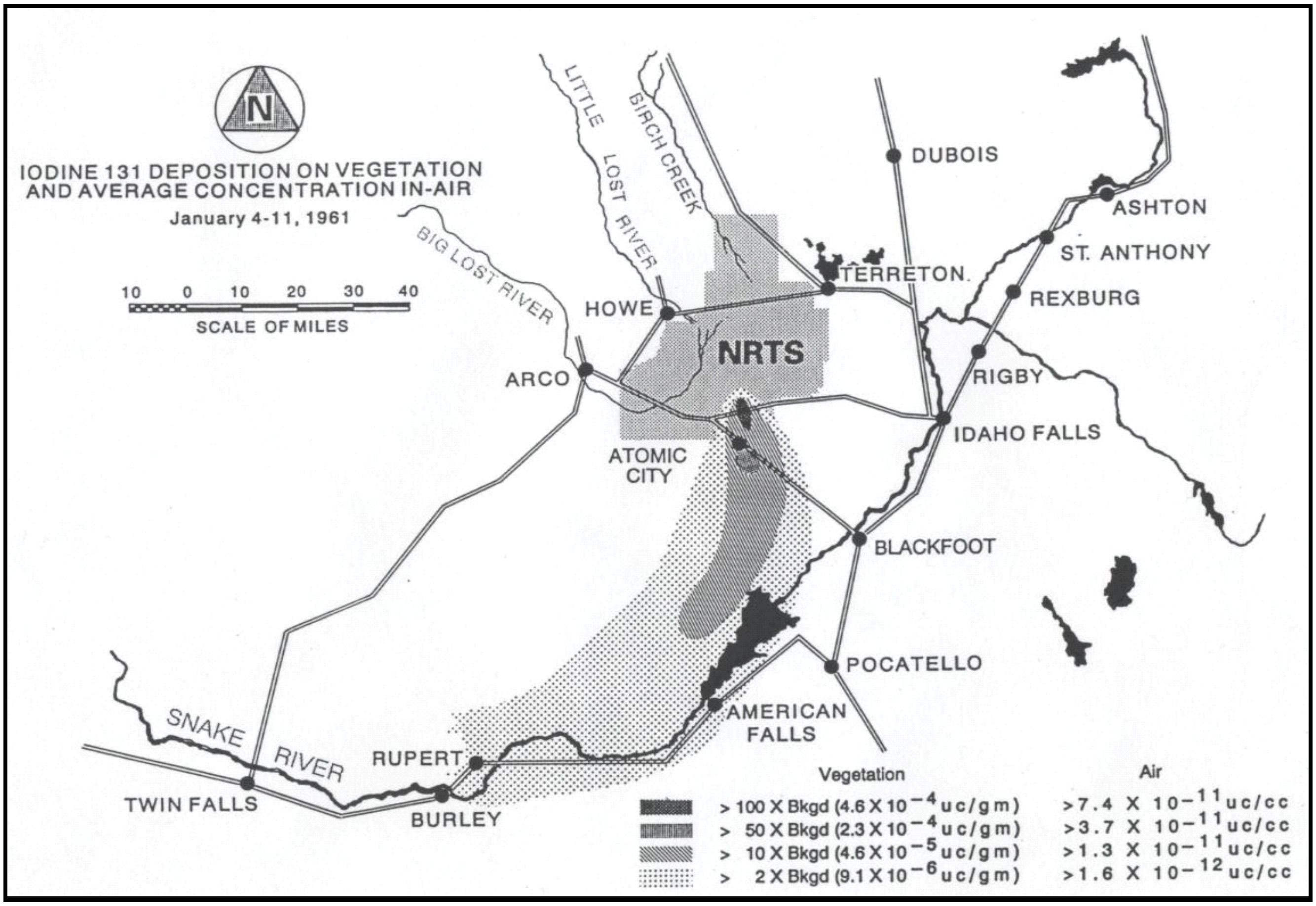

Figure 4-6. Dispersion Coefficient Contours for the SL-1 Accident (redrafted from Horan and Gammill, 1961).

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 the south end of the ICPP area and for the CFA. These concentrations and intake quantities would be applicable only if the individual was in the respective areas on the morning of October 16, 1959. The criticality occurred at approximately 3:00 a.m. Table 4-10 lists the intakes applicable at the southern end of ICPP and CFA.

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 x 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 the southern end of the ICPP and for the CFA. The source term used for this event is the same as that used for DOE (1991). Table 4-10 lists the intakes applicable at the southern end of ICPP and CFA, if the individual was in the respective areas on January 25, 1961.

4.2.2.2.3 ICPP Criticality of October 17, 1978

At approximately 8:40 p.m. on October 17, 1978, a criticality incident occurred in the first cycle uranium extraction system in the CPP-601 process building (DOE 1979; Casto 1980). The fissioning lasted for about 20 minutes and involved about 3.0×10^{18} fissions. This event did not result in significant radiation exposures to personnel, and there was no contamination of general plant areas. Releases to the environment which were filtered, consisted mainly of noble fission gases and small amounts of iodines. For this analysis, the release was treated as a 1-second release of noble fission gases and a small fraction of the halogens produced in the event. Table 4-10 lists the intakes applicable at the southern end of ICPP and CFA. These intake quantities would be applicable only if the individual was in the respective areas on the evening of October 17, 1978.

4.2.2.2.4 HTRE No. 3 Criticality Excursion (IET 13)

IET 13 was characterized as the critical experiments and low-power testing phase of the HTRE No. 3 reactor configuration for the GE-ANP Project. The low-power and critical experiments began on September 8 and ended on November 18, 1958, when a criticality excursion damaged every fuel element in the reactor core. As indicated in the GE literature concerning the test, the critical experiments and low-power testing of the assembly created insignificant radiological releases compared to the release resulting from the 770-megawatt-second energy excursion that ended this series of tests. More information concerning this incident is included in Peterson (2004).

The released material for this test is consistent with the release modeled in INELHDE (DOE 1991). Table 4-10 provides the resulting inhaled quantities that would be applicable at the LPTF/STPF location. The intakes are applicable only if the individual were at the Low Power Test Facility/Shield Test Pool Facility (LPTF/STPF) location on November 18, 1958.

4.2.2.3 Releases from Planned Tests

Of the 108 episodic releases analyzed in DOE (1991), only 16 had the potential to affect other INEEL facilities. Section 4.2.2.2 describes three of these events, the criticalities at ICPP and the critical excursion in the HTRE No.3. The 12 remaining events are:

1. Fuel Element Burn Test A
2. Fuel Element Burn Test B
3. Fuel Element Cutting Facility (FECF)
4. Initial Engine Test (IET) #14
5. IET 15(B)
6. IET 16
7. IET 17(b)
8. IET 18
9. IET 19(A)
10. IET 25(A)
11. IET 25(B)
12. IET 26(A)

For a given test, if an onsite facility existed between the point at which the test was conducted and the affected offsite location, that test was conservatively assumed to have affected an onsite facility or facilities. For example, the meteorological conditions that existed during Fuel Element Burn Test A, at the Grid III location, dictated that the offsite location of Reno Ranch, adjacent to the northern INEEL boundary as shown on Figure 4-1, be evaluated in DOE (1991). Inspection of the plume passage necessary to affect Reno Ranch indicates an impact at the TAN facility. The same conclusion for Fuel Element Burn Test B can be drawn, but in that case the plume impact on TAN facilities would be less direct because the offsite affected location was Birch Creek, which is considerably to the west of Reno Ranch. The Fuel Element Cutting Facility (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 CFA is in the straight-line path between ICPP and Frenchman's Cabin, a radiological impact analysis was conducted for the CFA.

Because all other test releases listed above, which originated at the TAN facility, affected two locations on the southern boundary [Frenchman's Cabin or Cerro Grande (shown on Figure 4-1), as evaluated in DOE (1991)], they might also have affected ICPP, TRA, SPERT, south INEEL Security Gate, or CFA. Because TRA and ICPP are essentially the same distance from TAN, concentrations at either location were assumed to be the same. The following sections discuss these events.

4.2.2.3.1 Fuel Element Burn Test A

This test supported the GE-ANP program, providing information on the hypothetical radiological release during the crash of a nuclear-powered plane (Brodsky and Beard 1960). This test, with a stainless-steel-clad fuel element that had been operated at 20 MW for 120 hours and decayed for 70 days, was conducted at Grid III at INEEL and began at 2:19 p.m. on March 20, 1957. The fuel element was assembled among other airframe parts to simulate an aircraft crash and burned for 2 hours using jet fuel as the burning agent. After 2 hours, the fuel element, which was intact, was believed to have released not much radioactive material. Of the two Fuel Element Burn Tests, Burn Test A probably impacted TAN more directly than Burn Test B. Table 4-11 lists the best estimates of the radionuclides and TAN intakes of those nuclides. These intakes would have been received only if the individual was at TAN on March 20, 1957.

4.2.2.3.2 Fuel Element Burn Test B

This test was conducted with a fuel element that had twice the curie content as the first burn test. The test began at 6:47 p.m. on the same day as the first burn test (March 20, 1957) and heated the fuel to much higher temperatures by supplying oxygen to the fire fueled by thermite, steel wool, and iron filings. The fuel element for this test underwent 6.16×10^{21} fissions and a subsequent decay of 250 days (Brodsky and Beard 1960, p. 37).

After 90 seconds, most of the fuel element had melted and dispersed. Figure 3-12 of DOE (1991) shows the trajectory of the plume, and Figure 3-13 shows the corresponding dispersion coefficient contours. Interpolation between the contours of 29 and $13{,}000 \times 10^{-14}$ h m^{-3} provided a concentration for the GE-ANP area in the range of 6×10^{-12} h m^{-3}, the value used for determining the dose of an individual at TAN. The radionuclides released and the respective curie values were the same as those for the DOE (1991) analysis. Table 4-11 lists the resulting intakes for TAN. These intakes would have been received only if the individual was at TAN on the evening of March 20, 1957.

4.2.2.3.3 Fuel Element Cutting Facility Filter Break

Fuel elements sent to ICPP contained structural components on the ends 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, decontamination operations were conducted 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 radioactive material 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 4-11 lists the best-estimate intakes of the radionuclides in the southern end of ICPP and the CFA 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.

4.2.2.3.4 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 4-12 lists the of the fission products released during the IET #14 test, and the intakes at TRA/ICPP and CFA. An individual would have been exposed to these concentrations only if present at these locations between April 24 and May 19, 1959. The above radiological doses are for a total exposure period of 26 days.

4.2.2.3.5 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, blisters were observed 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. 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 TRA/ICPP and CFA. Table 4-12 lists the release of fission products, which correspond with the Part B operation release documented in DOE (1991) for the intakes applicable at TRA/ICPP and CFA. An individual would only have been exposed to these concentrations and intakes if present at the locations between June 16 and June 24, 1959. The above radiological doses are for the total 9-day exposure period.

4.2.2.3.6 Initial Engine Test 16

This was the first power test conducted in HTRE No. 3 to determine the operating characteristics of the horizontal core. Because the operation was to determine these characteristics, most operations were at low power levels. The total operation occurred between July 28 and October 9, 1959, and produced only 95 MW-hours of power (Showalter 1959).

The fission product release for this test was modeled the same as that in DOE (1991) and was assumed to occur on October 9, 1959. Modeling releases for IET 16 involved the preservation of three factors: (1) a burnup of 95 MW-hours, (2) a conservative particulate release of 14 Ci (1.5 Ci h^{-1} for 9.5 hours), and (3) a conservative release fraction of 3.0×10^{-7} for iodine isotopes, the highest fraction measured during the test. To preserve these values and arrive at release fractions for other groups of radionuclides, engineering judgment and preliminary data from a few iterations of the RSAC program were used. To meet the stated criteria of 1.5 Ci h^{-1} for particulates, the noble gas release fraction was assumed to be 200 times greater than that for the iodines, and the release fraction for the solids was assumed to be 10% of the halogen release fraction.

The release for this test series was modeled the same as that for DOE (1991) with Cerro Grande, south of the INEEL boundary, as the location of highest offsite concentration. For this analysis, a straight-line trajectory from TAN to Cerro Grande intercepted the SPERT facility and the INEEL south security gate located just north of the junction of highways 26 and 20 on the road to CFA as shown on Figure 4-1. Table 4-12 lists the intakes at the respective locations.

4.2.2.3.7 Initial Engine Test 17(B)

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

4.2.2.3.8 Initial Engine Test 18

IET 18, conducted between December 23, 1959, and February 8, 1960, was designated as the Phase 2 testing of the HTRE No. 3 engine and was an extension of the test program outlined for IET 16. The following is a brief description of operations during this test series:

> *The powerplant was transported to Initial Engine Test on December 14, 1959, for final checkout in the facility prior to testing. The first engine operation was accomplished December 22, 1959, and the first data was taken December 23, 1959, (run No. I-1). The reactor was made critical on December 23, 1959, (run No. I-6).* (Highberg et al. 1960)
>
> *Damage was sustained to the instrumentation circuitry of the powerplant on January 6, 1960, as a result of failure of the electric aftercooling blowers. The powerplant was returned to the Hot Shop on January 7, 1960, for repairs that were completed January 12, 1960. Testing resumed at Initial Engine Test on January 21, 1960, and finished on February 7, 1960.*
>
> *The initial transfer from chemical operation to full nuclear operation occurred on January 26, 1960, (run No. 11-12), and the design conditions for endurance testing were initially attained at 11:58 p.m. on January 26, 1960 (run 11-32). A total of 126.42 hours of operation was achieved at design conditions with a continuous operation of 64.9 hours at these conditions. Operations were accomplished above 1% of design power for a total of 166.5 hours.*

With respect to effluent monitoring:

> *Continuous effluent monitoring was maintained to measure and to record the activity released to the atmosphere by the powerplant. The maximum output was 8.6 curies/hour (measured 10 minutes after release). The total output for the test series was 1157 curies (measured 10 minutes after release). The maximum release rate for I-131 was approximately 1.5 curies/hour (measured 10 minutes after release). The total offsite inhaled and ingested was below measurable amounts during this test series.* (Highberg et al. 1960).

The release for this test series was modeled the same as DOE (1991) with Cerro Grande, south of the INEEL boundary, as the location of highest concentration. For this analysis, a straight-line trajectory from TAN to Cerro Grande intercepted the SPERT facility and the INEEL south security gate. Table 4-12 lists the intakes for the two onsite locations.

4.2.2.3.9 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 4-12 lists intakes for TRA/ICPP and CFA. An individual would have been exposed to these intakes only if present at the respective locations between February 17 and February 29, 1960. The above radiological doses are for the total exposure period of 13 days.

4.2.2.3.10 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 TRA/ICPP and CFA. The intakes for this test are listed in Table 4-12. 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.

4.2.2.3.11 Initial Engine Test 25(B)

IET 25, performed between November 15 and December 19, 1960, was an extension of the Phase II testing program conducted in IET 18 and the second part of the IET 25 test (Linn 1962; Highberg et al. 1961). The releases for the test correspond with the significant periods of operation with IET 25(B) releases occurring from December 1 to December 15, 1960. Of the total release for IET 25, 76% was assumed to have been released during the 25(B) operation.

The releases for this test were the same as those modeled for DOE (1991) with a straight-line trajectory from TAN to SPERT and the INEEL south security gate. Table 4-12 lists the intakes corresponding to the two onsite locations. An individual would have received intakes only if present at the locations between December 1 and December 15, 1960. The above radiological intakes are for the total exposure period of 16 days.

4.2.2.3.12 Initial Engine Test 26(A)

IET 26, conducted in HTRE No. 2, was performed 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 being tested was the L2E-6 cartridge, which consisted of fueled and non-fueled, 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 TRA/ICPP and CFA. The intakes are listed in Table 4-12 for TRA/ICPP and CFA. 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 gaseous 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 Health & Safety (H&S) organization will observe and curb such a trend to prevent personnel exposures from increasing unnecessarily. An excellent example of another aspect that can cause facility background to increase is the operational tests that were conducted at the IET area at TAN. During an IET, such as one of those described in Section 4.2.2.3 where fuel damage created a significant release to the environment, the 76-inch duct from the HTRE engine to the stack was a very significant radiation source. That is why operational facilities for that area were heavily shielded and personnel were required to be within one of these shielded facilities during a test. The following sections discuss facility fence-line film badge and thermoluminescent dosimeter (TLD) data which recorded doses from gaseous fission product releases that had the potential for personnel exposure. More information on these two subjects is provided in Peterson (2004).

4.3.1 <u>Facility Fence-Line Annual Doses</u>

Before 1970, many film badge or TLD measurements were made inside the 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 increased to six weeks in 1962 and then changed back to monthly changes in 1963. Film badges were used up through 9 months of 1966 and TLDs were used from that time. Beginning in 1967 the TLDs were changed on a semi-annual 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. More information and film badge data for this early period are provided in Peterson (2004). The “detection limit” for the film badge reading was often quoted as 10 mrem for both a beta reading and a gamma reading (AEC 1963) and was quoted as being 10 mrem for the TLD when it was first used. With the annual background radiation field at the INEEL before operations started measured at 100 to 150 mrem/year, the monthly value of 8 to 13 mrem is at the detection limit of the film badge or the TLD. Therefore, the uncertainty for monthly changeouts is higher than for less frequent changeouts.

Between the latter part of 1970 and the latter part of 1972, facility fence monitoring and facility locations had been established. From 1972 through 1983, facility fence TLD measurements, made on a 6-month basis with 5 TLDs located at each facility position, are available in the Environmental Monitoring Data Reports (EMDRs) for the INEL. The imported table of Figure 4.7 (ERDA 1976) shows that uncertainty can vary from less than 10% to 20% for a given set of readings. At each of the 34 monitoring locations shown on this table, there are normally 5 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 set of readings are assigned a 2 sigma uncertainty of 16 to 20% and 18.4% of the readings are assigned a 2 sigma

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 4-7 Example of On-site TLD Monitoring Data.

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%.

To supply facility values for the 1952 to 1972 period, the highest 6-month value from 4/72 to 4/73 for the respective facility was multiplied by 2 and applied to each year between 1952 and 1972. (It should be remembered that not all of the listed facilities commenced operation or even existed in 1952.) Facility fence TLD measurements could not be located for 1984 through 1992, but for 1993 and beyond, such facility fence-line measurements are again included in the INEL/INEEL EMRs. For the 1984 to 1992 period where TLD measurements are missing, reasonable extrapolations are used to provide the missing values. In addition, background TLD measurements, corresponding with the facility fence TLD measuring periods, were recorded in the EMRs. All reduced facility fence-line TLD data (facility fence-line data minus background) in the EMRs are listed in Table 4-13. A more detailed discussion of the data is included in Peterson (2004).

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 presented in Table 4-13.

4.4 UNCERTAINTY

A detailed discussion of the derivation of airborne releases for operational conditions and episodic events is provided in INELHDE (DOE 1991).

Operational Releases - Discussions with the authors of the INELHDE 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 be increased. 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.

Episodic Releases - As described in INELHDE, 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, however, have stated that the release considered for a particular episodic event might be low by as much as a factor of 3.

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 of 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: a) 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, b) releases involving long half-lived, aged fission products (Fuel Element Burn Tests and the release from the Fuel Element Cutting Facility Filter Break), and c) 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.

During1988 a minor contamination incident occurred at the RWMC, which led to minor inhalation exposures to a few RWMC individuals, but that incident has not been incorporated into the environmental doses. The following paragraph, excerpted from the 1988 EMR (DOE 1989) summarizes site-monitoring information for this incident:

> *In early spring of 1988, waste boxes which had been stored on an outdoor asphalt pad at the RWMC (some since 1986) were to be moved indoors. A box which had been stored outside was discovered to be breached with some contamination spread. Further investigation of approximately 500 boxes on the pad found three other boxes that had been breached. Since these boxes had been stored on the pad for more than a year and were known to contain primarily Pu-238, it is now assumed that concentrations of that radionuclide reported on RWMC air filters in 1986 and 1988 were due to the small releases from those boxes. The highest RWMC concentration of Pu-238 in 1986 was 0.06% of the derived concentration guide and in 1988 was 0.04% of the guide. Cleanup and contamination controls were instituted at RWMC, and the entire stack of boxes was covered in November 1988 as an interim measure to control the spread of contamination prior to a permanent solution to the problem.*

Film Badge and TLD Measurements – As discussed in Section 4.3.1, above, the uncertainty of individual measurements, made with film badges and TLDs, can be as high as $\pm$ 100%, depending on the frequency of changeout, i.e., once per month, which was generally the case with film badges. The data for 1952 to 1972 of Table 4-13 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 50s and early 60s (the last IET test, IET #26, ended on March 31, 1961), it should be remembered that 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 impacted. In spite of these controls, the 1952 to 1962 values for the TAN areas (TAN-TSF, TAN-LOFT, and TAN-LPT) could be low by a factor of 3. However, after 1967 when facility fence-line measurements were routinely made with TLDs, with 5 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%.

Dose reconstruction for individuals whose location is unknown should use intakes provided by Table 4-3 (CFA) and exposures for ICPP as provided in Table 4-13. The values suggested will maximize the resultant individual dose.

Table 4-1: Intake (Bq yr^{-1}) by Year for ANL 1952-2002.

Nuclide	Ce-144	I-131	Pm-147	Pu-238	Pu-239, 240	Ru-106	Sr-89	Sr-90	Y-91
1952	5.5E+0	3.3E-2	0.0E+0	0.0E+0	0.0E+0	4.1E-1	0.0E+0	1.3E-1	5.0E-1
1953	5.5E+0	7.3E-2	1.3E+0	8.9E-4	1.3E-4	4.1E-1	4.0E-1	6.8E-1	8.6E-1
1954	1.4E+1	5.3E-2	3.3E+0	2.2E-3	3.3E-4	1.0E+0	9.9E-1	1.2E+0	2.1E+0
1955	1.9E+1	7.9E-2	4.7E+0	2.2E-3	3.3E-4	1.4E+0	1.4E+0	1.7E+0	3.0E+0
1956	2.2E+1	9.4E-1	5.4E+0	3.6E-3	5.3E-4	1.6E+0	1.7E+0	1.8E+0	3.5E+0
1957	4.4E+0	1.1E+2	1.1E+1	6.3E-3	9.3E-4	4.7E-1	1.6E+0	2.9E+0	1.7E+0
1958	6.0E+0	8.3E+1	1.6E+1	9.1E-3	1.3E-3	6.6E-1	1.2E+0	4.1E+0	1.3E+0
1959	4.8E+0	1.8E+1	1.3E+1	7.3E-3	1.1E-3	5.4E-1	6.1E-1	3.4E+0	6.7E-1
1960	6.6E-2	2.6E+0	1.4E-1	2.5E-3	3.7E-4	6.6E-3	7.6E-2	2.1E-1	8.2E-2
1961	4.6E-2	3.5E+0	3.3E-3	4.6E-4	6.8E-5	2.8E-3	2.0E-1	2.5E-1	2.1E-1
1962	2.1E-1	3.3E+0	4.6E-1	3.0E-4	4.5E-5	2.1E-2	2.0E-1	3.8E-1	2.2E-1
1963	3.3E+0	2.1E+0	9.4E+0	2.8E-3	4.2E-4	3.7E-1	1.3E-1	2.8E+0	1.4E-1
1964	1.8E+0	1.1E-1	0.0E+0	1.1E-4	1.6E-5	2.8E+1	3.0E-2	7.1E-1	2.0E+0
1965	4.6E+0	7.4E-1	0.0E+0	4.7E-3	7.0E-4	2.0E+0	0.0E+0	2.7E+0	1.8E+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

Table 4-2: Intake (Bq yr^{-1}) by Year for ARA 1952-2002

Nuclide	Ce-144	I-131	Pm-147	Pu-238	Pu-239, 240	Ru-106	Sr-89	Sr-90	Y-91
1952	7.9E+0	4.7E-2	0.0E+0	0.0E+0	0.0E+0	5.8E-1	0.0E+0	1.8E-1	7.2E-1
1953	7.9E+0	1.0E-1	1.9E+0	1.3E-3	1.9E-4	5.8E-1	5.7E-1	9.7E-1	1.2E+0
1954	2.0E+1	7.6E-2	4.8E+0	3.2E-3	4.7E-4	1.5E+0	1.4E+0	1.7E+0	3.1E+0
1955	2.8E+1	1.1E-1	6.7E+0	3.2E-3	4.7E-4	2.0E+0	2.0E+0	2.4E+0	4.3E+0
1956	3.2E+1	1.3E+0	7.7E+0	5.1E-3	7.5E-4	2.3E+0	2.4E+0	2.5E+0	5.0E+0
1957	6.2E+0	1.6E+2	1.6E+1	9.0E-3	1.3E-3	6.8E-1	2.2E+0	4.1E+0	2.4E+0
1958	8.5E+0	1.2E+2	2.3E+1	1.3E-2	1.9E-3	9.4E-1	1.7E+0	5.9E+0	1.9E+0
1959	6.9E+0	2.6E+1	1.9E+1	1.0E-2	1.5E-3	7.7E-1	8.7E-1	4.9E+0	9.6E-1
1960	9.5E-2	3.8E+0	2.0E-1	3.6E-3	5.3E-4	9.4E-3	1.1E-1	3.0E-1	1.2E-1
1961	6.6E-2	4.9E+0	4.7E-3	6.6E-4	9.7E-5	3.9E-3	2.8E-1	3.6E-1	3.1E-1
1962	3.0E-1	4.7E+0	6.5E-1	4.3E-4	6.4E-5	3.0E-2	2.9E-1	5.4E-1	3.1E-1
1963	4.8E+0	3.1E+0	1.3E+1	4.1E-3	5.9E-4	5.4E-1	1.8E-1	4.0E+0	2.0E-1
1964	2.6E+0	1.6E-1	0.0E+0	1.6E-4	2.3E-5	3.9E+1	4.3E-2	1.0E+0	2.8E+0
1965	6.5E+0	1.1E+0	0.0E+0	6.8E-3	1.0E-3	2.8E+0	0.0E+0	3.9E+0	2.5E+0
1966	3.9E+0	6.2E-1	0.0E+0	1.5E-3	2.3E-4	1.8E+1	0.0E+0	1.1E+0	1.7E+0
1967	9.9E-2	2.6E-1	0.0E+0	1.7E-4	2.5E-5	2.5E+0	0.0E+0	3.0E-1	9.5E-1
1968	7.1E+0	4.8E-1	0.0E+0	3.3E-3	4.8E-4	1.1E+0	0.0E+0	1.7E+0	9.1E-1
1969	4.0E-1	6.9E-1	0.0E+0	7.2E-4	1.1E-4	5.3E-1	0.0E+0	5.1E-1	7.6E-1
1970	9.5E-1	3.6E-5	0.0E+0	1.0E-3	1.5E-4	4.3E-1	0.0E+0	3.8E-1	7.8E-1
1971	3.6E+0	9.9E-1	0.0E+0	3.0E-3	4.4E-4	4.6E+0	0.0E+0	1.6E+0	6.6E-1
1972	3.9E-1	4.2E-1	0.0E+0	9.2E-4	1.4E-4	7.0E-1	0.0E+0	4.0E-1	2.4E-1
1973	8.1E-3	4.6E-6	0.0E+0	8.0E-5	1.2E-5	3.8E-2	0.0E+0	2.1E-2	7.3E-3
1974	2.4E-2	3.8E-3	0.0E+0	3.3E-4	2.9E-5	1.3E-1	0.0E+0	1.1E-1	3.7E-1
1975	1.3E-2	6.5E-3	0.0E+0	1.8E-4	3.6E-5	9.1E-2	0.0E+0	2.8E-2	3.5E-1
1976	7.4E-5	4.6E-5	0.0E+0	1.2E-5	5.1E-6	1.2E-3	0.0E+0	5.5E-4	4.3E-2
1977	2.9E-4	2.0E-4	0.0E+0	1.1E-4	4.8E-5	1.5E-2	0.0E+0	7.2E-3	6.1E-1
1978	5.4E-4	2.9E-3	0.0E+0	1.1E-4	1.1E-5	8.1E-3	0.0E+0	2.8E-3	5.0E-1
1979	7.6E-4	4.2E-4	0.0E+0	2.0E-4	2.2E-5	5.5E-3	0.0E+0	3.8E-2	2.2E-1
1980	1.2E-3	6.2E-3	0.0E+0	1.3E-4	1.7E-5	2.7E-3	0.0E+0	1.8E-3	1.3E+0
1981	1.2E-3	1.6E-2	0.0E+0	2.6E-5	4.8E-6	2.6E-2	0.0E+0	1.4E-3	8.6E-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	4.1E-4	2.1E-3	0.0E+0	1.7E-4	2.3E-5	3.3E-3	0.0E+0	1.5E-4	5.2E-2
1984	4.1E-4	1.4E-4	0.0E+0	2.7E-5	1.1E-5	4.5E-4	0.0E+0	1.8E-4	2.1E-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	1.2E-3	3.8E-4	0.0E+0	5.5E-6	4.2E-7	1.0E-2	0.0E+0	6.9E-5	1.9E-1
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	1.2E-3	4.2E-5	0.0E+0	2.5E-8	3.5E-9	6.9E-4	0.0E+0	3.2E-5	2.9E-1
1990	4.1E-4	1.3E-4	0.0E+0	3.1E-9	3.1E-9	1.4E-4	0.0E+0	6.6E-7	1.0E-1
1991	4.1E-4	5.2E-5	0.0E+0	3.3E-10	3.3E-10	1.7E-4	0.0E+0	3.2E-4	6.0E-2
1992	4.1E-4	6.0E-5	0.0E+0	5.7E-7	5.7E-7	4.6E-5	0.0E+0	2.8E-5	1.0E-1
1993	0.0E+0	3.8E-5	0.0E+0	0.0E+0	8.7E-10	3.5E-4	0.0E+0	3.8E-4	0.0E+0
1994	0.0E+0	1.3E-4	0.0E+0	0.0E+0	2.0E-7	0.0E+0	0.0E+0	3.4E-4	0.0E+0
1995	0.0E+0	4.8E-5	0.0E+0	7.7E-8	1.3E-8	0.0E+0	0.0E+0	5.4E-6	0.0E+0
1996	0.0E+0	6.5E-5	0.0E+0	5.1E-7	1.0E-8	0.0E+0	0.0E+0	2.5E-6	0.0E+0
1997	0.0E+0	0.0E+0	0.0E+0	4.1E-7	1.3E-7	0.0E+0	0.0E+0	5.7E-5	0.0E+0
1998	0.0E+0	5.4E-5	0.0E+0	4.0E-7	4.3E-8	0.0E+0	0.0E+0	2.5E-5	0.0E+0
1999	0.0E+0	1.0E-4	0.0E+0	2.5E-7	2.4E-8	0.0E+0	0.0E+0	1.5E-5	0.0E+0
2000	0.0E+0	6.4E-3	0.0E+0	1.2E-4	1.2E-6	0.0E+0	0.0E+0	1.2E-2	0.0E+0
2001	0.0E+0	3.3E-3	0.0E+0	8.0E-7	9.5E-5	0.0E+0	0.0E+0	3.9E-4	0.0E+0
2002	0.0E+0	3.7E-4	0.0E+0	1.6E-5	4.2E-5	0.0E+0	0.0E+0	8.9E-3	0.0E+0

Table 4-3: Intake (Bq yr^{-1}) by Year for CFA 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	1.2E+1	1.9E+0	0.0E+0	4.6E-3	6.8E-4	5.3E+1	0.0E+0	3.3E+0	5.0E+0
1967	3.0E-1	7.9E-1	0.0E+0	5.2E-4	7.6E-5	7.4E+0	0.0E+0	9.0E-1	2.8E+0
1968	2.1E+1	1.4E+0	0.0E+0	1.0E-2	1.5E-3	3.2E+0	0.0E+0	5.2E+0	2.7E+0
1969	1.2E+0	2.1E+0	0.0E+0	2.1E-3	3.2E-4	1.6E+0	0.0E+0	1.5E+0	2.3E+0
1970	2.8E+0	1.1E-4	0.0E+0	3.0E-3	4.5E-4	1.3E+0	0.0E+0	1.1E+0	2.3E+0
1971	1.1E+1	3.0E+0	0.0E+0	9.0E-3	1.3E-3	1.4E+1	0.0E+0	4.8E+0	2.0E+0
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	5.3E-4	0.0E+0	2.2E-5	6.0E-5	0.0E+0	0.0E+0	1.3E-2	0.0E+0

Table 4-4: Intake (Bq yr^{-1}) by Year for ICPP 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	4.0E-1	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	9.9E-1	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	1.4E+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	1.7E+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	1.6E+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	1.2E+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	6.1E-1	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	7.6E-2	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	2.0E-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	2.0E-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	1.3E-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	3.0E-2	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	1.2E+1	1.9E+0	0.0E+0	4.6E-3	6.8E-4	5.3E+1	0.0E+0	3.3E+0	5.0E+0
1967	3.0E-1	7.9E-1	0.0E+0	5.2E-4	7.6E-5	7.4E+0	0.0E+0	9.0E-1	2.8E+0
1968	2.1E+1	1.4E+0	0.0E+0	1.0E-2	1.5E-3	3.2E+0	0.0E+0	5.2E+0	2.7E+0
1969	1.2E+0	2.1E+0	0.0E+0	2.1E-3	3.2E-4	1.6E+0	0.0E+0	1.5E+0	2.3E+0
1970	2.8E+0	1.1E-4	0.0E+0	3.0E-3	4.5E-4	1.3E+0	0.0E+0	1.1E+0	2.3E+0
1971	1.1E+1	3.0E+0	0.0E+0	9.0E-3	1.3E-3	1.4E+1	0.0E+0	4.8E+0	2.0E+0
1972	1.2E+0	1.2E+0	0.0E+0	2.8E-3	4.2E-4	2.1E+0	0.0E+0	1.2E+0	7.2E-1
1973	5.7E-2	3.2E-5	0.0E+0	5.6E-4	8.1E-5	2.7E-1	0.0E+0	1.5E-1	5.1E-2
1974	7.9E-2	1.3E-2	0.0E+0	1.1E-3	9.7E-5	4.4E-1	0.0E+0	3.7E-1	1.2E+0
1975	3.8E-2	1.9E-2	0.0E+0	5.5E-4	1.1E-4	2.7E-1	0.0E+0	8.3E-2	1.1E+0
1976	5.2E-4	3.2E-4	0.0E+0	8.1E-5	3.6E-5	8.1E-3	0.0E+0	3.9E-3	3.0E-1
1977	8.7E-4	5.9E-4	0.0E+0	3.4E-4	1.5E-4	4.5E-2	0.0E+0	2.1E-2	1.8E+0
1978	1.6E-3	8.7E-3	0.0E+0	3.2E-4	3.4E-5	2.4E-2	0.0E+0	8.3E-3	1.5E+0
1979	7.6E-4	4.2E-4	0.0E+0	2.0E-4	2.2E-5	5.5E-3	0.0E+0	3.8E-2	2.2E-1
1980	1.2E-3	6.2E-3	0.0E+0	1.3E-4	1.7E-5	2.7E-3	0.0E+0	1.8E-3	1.3E+0
1981	1.2E-3	1.6E-2	0.0E+0	2.6E-5	4.8E-6	2.6E-2	0.0E+0	1.4E-3	8.6E-1
1982	2.2E-4	6.7E-5	0.0E+0	2.1E-5	2.3E-6	6.3E-4	0.0E+0	4.0E-4	1.0E-1
1983	1.2E-3	6.2E-3	0.0E+0	5.2E-4	6.9E-5	1.0E-2	0.0E+0	4.5E-4	1.5E-1
1984	1.2E-3	4.2E-4	0.0E+0	8.0E-5	3.2E-5	1.4E-3	0.0E+0	5.5E-4	6.2E-2
1985	2.9E-3	2.1E-2	0.0E+0	5.3E-5	1.1E-5	2.3E-2	0.0E+0	1.5E-3	2.3E+0
1986	1.2E-3	3.8E-4	0.0E+0	5.5E-6	4.2E-7	1.0E-2	0.0E+0	6.9E-5	1.9E-1
1987	1.2E-3	1.8E-4	0.0E+0	5.9E-6	9.0E-7	1.3E-4	0.0E+0	9.7E-5	3.3E+0
1988	1.2E-3	5.9E-5	0.0E+0	4.8E-6	7.3E-7	6.6E-2	0.0E+0	1.2E-4	2.2E+0
1989	1.2E-3	4.2E-5	0.0E+0	2.5E-8	3.5E-9	6.9E-4	0.0E+0	3.2E-5	2.9E-1
1990	4.1E-4	1.3E-4	0.0E+0	3.1E-9	3.1E-9	1.4E-4	0.0E+0	6.6E-7	1.0E-1
1991	4.1E-4	5.2E-5	0.0E+0	3.3E-10	3.3E-10	1.7E-4	0.0E+0	3.2E-4	6.0E-2
1992	1.2E-3	1.8E-4	0.0E+0	1.7E-6	1.7E-6	1.4E-4	0.0E+0	8.3E-5	3.1E-1
1993	0.0E+0	3.8E-5	0.0E+0	0.0E+0	8.7E-10	3.5E-4	0.0E+0	3.8E-4	0.0E+0
1994	0.0E+0	1.3E-4	0.0E+0	0.0E+0	2.0E-7	0.0E+0	0.0E+0	3.4E-4	0.0E+0
1995	0.0E+0	6.9E-5	0.0E+0	1.1E-7	1.8E-8	0.0E+0	0.0E+0	7.7E-6	0.0E+0
1996	0.0E+0	9.2E-5	0.0E+0	7.3E-7	1.5E-8	0.0E+0	0.0E+0	3.6E-6	0.0E+0
1997	0.0E+0	0.0E+0	0.0E+0	5.9E-7	1.8E-7	0.0E+0	0.0E+0	8.1E-5	0.0E+0
1998	0.0E+0	2.3E-4	0.0E+0	1.7E-6	1.8E-7	0.0E+0	0.0E+0	1.1E-4	0.0E+0
1999	0.0E+0	3.1E-4	0.0E+0	7.5E-7	7.1E-8	0.0E+0	0.0E+0	4.4E-5	0.0E+0
2000	0.0E+0	6.4E-3	0.0E+0	1.2E-4	1.2E-6	0.0E+0	0.0E+0	1.2E-2	0.0E+0
2001	0.0E+0	3.3E-3	0.0E+0	8.0E-7	9.5E-5	0.0E+0	0.0E+0	3.9E-4	0.0E+0
2002	0.0E+0	1.6E-3	0.0E+0	6.7E-5	1.8E-4	0.0E+0	0.0E+0	3.8E-2	0.0E+0

Table 4-5: Intake (Bq yr^{-1}) by Year for RWMC 1952-2002.

Nuclide	Ce-144	I-131	Pm-147	Pu-238	Pu-239, 240	Ru-106	Sr-89	Sr-90	Y-91
1952	5.5E+0	3.3E-2	0.0E+0	0.0E+0	0.0E+0	4.1E-1	0.0E+0	1.3E-1	5.0E-1
1953	5.5E+0	7.3E-2	1.3E+0	8.9E-4	1.3E-4	4.1E-1	4.0E-1	6.8E-1	8.6E-1
1954	1.4E+1	5.3E-2	3.3E+0	2.2E-3	3.3E-4	1.0E+0	9.9E-1	1.2E+0	2.1E+0
1955	1.9E+1	7.9E-2	4.7E+0	2.2E-3	3.3E-4	1.4E+0	1.4E+0	1.7E+0	3.0E+0
1956	2.2E+1	9.4E-1	5.4E+0	3.6E-3	5.3E-4	1.6E+0	1.7E+0	1.8E+0	3.5E+0
1957	4.4E+0	1.1E+2	1.1E+1	6.3E-3	9.3E-4	4.7E-1	1.6E+0	2.9E+0	1.7E+0
1958	6.0E+0	8.3E+1	1.6E+1	9.1E-3	1.3E-3	6.6E-1	1.2E+0	4.1E+0	1.3E+0
1959	4.8E+0	1.8E+1	1.3E+1	7.3E-3	1.1E-3	5.4E-1	6.1E-1	3.4E+0	6.7E-1
1960	6.6E-2	2.6E+0	1.4E-1	2.5E-3	3.7E-4	6.6E-3	7.6E-2	2.1E-1	8.2E-2
1961	4.6E-2	3.5E+0	3.3E-3	4.6E-4	6.8E-5	2.8E-3	2.0E-1	2.5E-1	2.1E-1
1962	2.1E-1	3.3E+0	4.6E-1	3.0E-4	4.5E-5	2.1E-2	2.0E-1	3.8E-1	2.2E-1
1963	3.3E+0	2.1E+0	9.4E+0	2.8E-3	4.2E-4	3.7E-1	1.3E-1	2.8E+0	1.4E-1
1964	1.8E+0	1.1E-1	0.0E+0	1.1E-4	1.6E-5	2.8E+1	3.0E-2	7.1E-1	2.0E+0
1965	4.6E+0	7.4E-1	0.0E+0	4.7E-3	7.0E-4	2.0E+0	0.0E+0	2.7E+0	1.8E+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	5.7E-2	3.2E-5	0.0E+0	5.6E-4	8.1E-5	2.7E-1	0.0E+0	1.5E-1	5.1E-2
1974	2.4E-2	3.8E-3	0.0E+0	3.3E-4	2.9E-5	1.3E-1	0.0E+0	1.1E-1	3.7E-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	7.4E-5	4.6E-5	0.0E+0	1.2E-5	5.1E-6	1.2E-3	0.0E+0	5.5E-4	4.3E-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	4.1E-4	1.4E-4	0.0E+0	2.7E-5	1.1E-5	4.5E-4	0.0E+0	1.8E-4	2.1E-2
1985	4.1E-4	3.0E-3	0.0E+0	7.6E-6	1.5E-6	3.3E-3	0.0E+0	2.2E-4	3.3E-1
1986	4.1E-4	1.3E-4	0.0E+0	1.8E-6	1.4E-7	3.3E-3	0.0E+0	2.3E-5	6.2E-2
1987	4.1E-4	6.1E-5	0.0E+0	2.0E-6	3.0E-7	4.3E-5	0.0E+0	3.2E-5	1.1E+0
1988	1.2E-3	5.9E-5	0.0E+0	4.8E-6	7.3E-7	6.6E-2	0.0E+0	1.2E-4	2.2E+0
1989	4.1E-4	1.4E-5	0.0E+0	8.2E-9	1.2E-9	2.3E-4	0.0E+0	1.1E-5	9.6E-2
1990	4.1E-4	1.3E-4	0.0E+0	3.1E-9	3.1E-9	1.4E-4	0.0E+0	6.6E-7	1.0E-1
1991	4.1E-4	5.2E-5	0.0E+0	3.3E-10	3.3E-10	1.7E-4	0.0E+0	3.2E-4	6.0E-2
1992	4.1E-4	6.0E-5	0.0E+0	5.7E-7	5.7E-7	4.6E-5	0.0E+0	2.8E-5	1.0E-1
1993	0.0E+0	3.8E-5	0.0E+0	0.0E+0	8.7E-10	3.5E-4	0.0E+0	3.8E-4	0.0E+0
1994	0.0E+0	1.3E-4	0.0E+0	0.0E+0	2.0E-7	0.0E+0	0.0E+0	3.4E-4	0.0E+0
1995	0.0E+0	6.9E-5	0.0E+0	1.1E-7	1.8E-8	0.0E+0	0.0E+0	7.7E-6	0.0E+0
1996	0.0E+0	6.5E-5	0.0E+0	5.1E-7	1.0E-8	0.0E+0	0.0E+0	2.5E-6	0.0E+0
1997	0.0E+0	0.0E+0	0.0E+0	4.1E-7	1.3E-7	0.0E+0	0.0E+0	5.7E-5	0.0E+0
1998	0.0E+0	5.4E-5	0.0E+0	4.0E-7	4.3E-8	0.0E+0	0.0E+0	2.5E-5	0.0E+0
1999	0.0E+0	7.2E-5	0.0E+0	1.8E-7	1.7E-8	0.0E+0	0.0E+0	1.0E-5	0.0E+0
2000	0.0E+0	6.4E-3	0.0E+0	1.2E-4	1.2E-6	0.0E+0	0.0E+0	1.2E-2	0.0E+0
2001	0.0E+0	3.3E-3	0.0E+0	8.0E-7	9.5E-5	0.0E+0	0.0E+0	3.9E-4	0.0E+0
2002	0.0E+0	5.3E-4	0.0E+0	2.2E-5	6.0E-5	0.0E+0	0.0E+0	1.3E-2	0.0E+0

Table 4-6: Intake (Bq yr^{-1}) by Year for SPERT 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	1.2E+1	1.9E+0	0.0E+0	4.6E-3	6.8E-4	5.3E+1	0.0E+0	3.3E+0	5.0E+0
1967	3.0E-1	7.9E-1	0.0E+0	5.2E-4	7.6E-5	7.4E+0	0.0E+0	9.0E-1	2.8E+0
1968	2.1E+1	1.4E+0	0.0E+0	1.0E-2	1.5E-3	3.2E+0	0.0E+0	5.2E+0	2.7E+0
1969	1.2E+0	2.1E+0	0.0E+0	2.1E-3	3.2E-4	1.6E+0	0.0E+0	1.5E+0	2.3E+0
1970	2.8E+0	1.1E-4	0.0E+0	3.0E-3	4.5E-4	1.3E+0	0.0E+0	1.1E+0	2.3E+0
1971	1.1E+1	3.0E+0	0.0E+0	9.0E-3	1.3E-3	1.4E+1	0.0E+0	4.8E+0	2.0E+0
1972	1.2E+0	1.2E+0	0.0E+0	2.8E-3	4.2E-4	2.1E+0	0.0E+0	1.2E+0	7.2E-1
1973	8.1E-3	4.6E-6	0.0E+0	8.0E-5	1.2E-5	3.8E-2	0.0E+0	2.1E-2	7.3E-3
1974	5.5E-2	8.9E-3	0.0E+0	7.8E-4	6.8E-5	3.1E-1	0.0E+0	2.6E-1	8.6E-1
1975	3.8E-2	1.9E-2	0.0E+0	5.5E-4	1.1E-4	2.7E-1	0.0E+0	8.3E-2	1.1E+0
1976	7.4E-4	4.6E-4	0.0E+0	1.2E-4	5.1E-5	1.2E-2	0.0E+0	5.5E-3	4.3E-1
1977	8.7E-4	5.9E-4	0.0E+0	3.4E-4	1.5E-4	4.5E-2	0.0E+0	2.1E-2	1.8E+0
1978	1.6E-3	8.7E-3	0.0E+0	3.2E-4	3.4E-5	2.4E-2	0.0E+0	8.3E-3	1.5E+0
1979	2.5E-3	1.4E-3	0.0E+0	6.8E-4	7.4E-5	1.8E-2	0.0E+0	1.3E-1	7.3E-1
1980	1.2E-3	6.2E-3	0.0E+0	1.3E-4	1.7E-5	2.7E-3	0.0E+0	1.8E-3	1.3E+0
1981	1.2E-3	1.6E-2	0.0E+0	2.6E-5	4.8E-6	2.6E-2	0.0E+0	1.4E-3	8.6E-1
1982	2.2E-4	6.7E-5	0.0E+0	2.1E-5	2.3E-6	6.3E-4	0.0E+0	4.0E-4	1.0E-1
1983	1.2E-3	6.2E-3	0.0E+0	5.2E-4	6.9E-5	1.0E-2	0.0E+0	4.5E-4	1.5E-1
1984	1.2E-3	4.2E-4	0.0E+0	8.0E-5	3.2E-5	1.4E-3	0.0E+0	5.5E-4	6.2E-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	1.2E-3	3.8E-4	0.0E+0	5.5E-6	4.2E-7	1.0E-2	0.0E+0	6.9E-5	1.9E-1
1987	1.2E-3	1.8E-4	0.0E+0	5.9E-6	9.0E-7	1.3E-4	0.0E+0	9.7E-5	3.3E+0
1988	4.1E-4	2.0E-5	0.0E+0	1.6E-6	2.4E-7	2.2E-2	0.0E+0	3.9E-5	7.4E-1
1989	1.2E-3	4.2E-5	0.0E+0	2.5E-8	3.5E-9	6.9E-4	0.0E+0	3.2E-5	2.9E-1
1990	4.1E-4	1.3E-4	0.0E+0	3.1E-9	3.1E-9	1.4E-4	0.0E+0	6.6E-7	1.0E-1
1991	4.1E-4	5.2E-5	0.0E+0	3.3E-10	3.3E-10	1.7E-4	0.0E+0	3.2E-4	6.0E-2
1992	1.2E-3	1.8E-4	0.0E+0	1.7E-6	1.7E-6	1.4E-4	0.0E+0	8.3E-5	3.1E-1
1993	0.0E+0	3.8E-5	0.0E+0	0.0E+0	8.7E-10	3.5E-4	0.0E+0	3.8E-4	0.0E+0
1994	0.0E+0	1.3E-4	0.0E+0	0.0E+0	2.0E-7	0.0E+0	0.0E+0	3.4E-4	0.0E+0
1995	0.0E+0	6.9E-5	0.0E+0	1.1E-7	1.8E-8	0.0E+0	0.0E+0	7.7E-6	0.0E+0
1996	0.0E+0	6.5E-5	0.0E+0	5.1E-7	1.0E-8	0.0E+0	0.0E+0	2.5E-6	0.0E+0
1997	0.0E+0	0.0E+0	0.0E+0	4.1E-7	1.3E-7	0.0E+0	0.0E+0	5.7E-5	0.0E+0
1998	0.0E+0	5.4E-5	0.0E+0	4.0E-7	4.3E-8	0.0E+0	0.0E+0	2.5E-5	0.0E+0
1999	0.0E+0	3.1E-4	0.0E+0	7.5E-7	7.1E-8	0.0E+0	0.0E+0	4.4E-5	0.0E+0
2000	0.0E+0	6.4E-3	0.0E+0	1.2E-4	1.2E-6	0.0E+0	0.0E+0	1.2E-2	0.0E+0
2001	0.0E+0	3.3E-3	0.0E+0	8.0E-7	9.5E-5	0.0E+0	0.0E+0	3.9E-4	0.0E+0
2002	0.0E+0	3.7E-4	0.0E+0	1.6E-5	4.2E-5	0.0E+0	0.0E+0	8.9E-3	0.0E+0

Table 4-7. Intake (Bq yr^{-1}) by Year for TAN 1952-2002.

Nuclide	Ce-144	I-131	Pm-147	Pu-238	Pu-239, 240	Ru-106	Sr-89	Sr-90	Y-91
1952	5.5E+0	3.3E-2	0.0E+0	0.0E+0	0.0E+0	4.1E-1	0.0E+0	1.3E-1	5.0E-1
1953	5.5E+0	7.3E-2	1.3E+0	8.9E-4	1.3E-4	4.1E-1	4.0E-1	6.8E-1	8.6E-1
1954	1.4E+1	5.3E-2	3.3E+0	2.2E-3	3.3E-4	1.0E+0	9.9E-1	1.2E+0	2.1E+0
1955	1.9E+1	7.9E-2	4.7E+0	2.2E-3	3.3E-4	1.4E+0	1.4E+0	1.7E+0	3.0E+0
1956	2.2E+1	9.4E-1	5.4E+0	3.6E-3	5.3E-4	1.6E+0	1.7E+0	1.8E+0	3.5E+0
1957	4.4E+0	1.1E+2	1.1E+1	6.3E-3	9.3E-4	4.7E-1	1.6E+0	2.9E+0	1.7E+0
1958	6.0E+0	8.3E+1	1.6E+1	9.1E-3	1.3E-3	6.6E-1	1.2E+0	4.1E+0	1.3E+0
1959	4.8E+0	1.8E+1	1.3E+1	7.3E-3	1.1E-3	5.4E-1	6.1E-1	3.4E+0	6.7E-1
1960	6.6E-2	2.6E+0	1.4E-1	2.5E-3	3.7E-4	6.6E-3	7.6E-2	2.1E-1	8.2E-2
1961	4.6E-2	3.5E+0	3.3E-3	4.6E-4	6.8E-5	2.8E-3	2.0E-1	2.5E-1	2.1E-1
1962	2.1E-1	3.3E+0	4.6E-1	3.0E-4	4.5E-5	2.1E-2	2.0E-1	3.8E-1	2.2E-1
1963	3.3E+0	2.1E+0	9.4E+0	2.8E-3	4.2E-4	3.7E-1	1.3E-1	2.8E+0	1.4E-1
1964	1.8E+0	1.1E-1	0.0E+0	1.1E-4	1.6E-5	2.8E+1	3.0E-2	7.1E-1	2.0E+0
1965	4.6E+0	7.4E-1	0.0E+0	4.7E-3	7.0E-4	2.0E+0	0.0E+0	2.7E+0	1.8E+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	3.8E-3	1.9E-3	0.0E+0	5.5E-5	1.1E-5	2.7E-2	0.0E+0	8.3E-3	1.1E-1
1976	7.4E-5	4.6E-5	0.0E+0	1.2E-5	5.1E-6	1.2E-3	0.0E+0	5.5E-4	4.3E-2
1977	2.9E-5	2.0E-5	0.0E+0	1.1E-5	4.8E-6	1.5E-3	0.0E+0	7.2E-4	6.1E-2
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	7.6E-5	4.2E-5	0.0E+0	2.0E-5	2.2E-6	5.5E-4	0.0E+0	3.8E-3	2.2E-2
1980	1.2E-4	6.2E-4	0.0E+0	1.3E-5	1.7E-6	2.7E-4	0.0E+0	1.8E-4	1.3E-1
1981	1.2E-4	1.6E-3	0.0E+0	2.6E-6	4.8E-7	2.6E-3	0.0E+0	1.4E-4	8.6E-2
1982	6.6E-5	2.0E-5	0.0E+0	6.2E-6	6.9E-7	1.9E-4	0.0E+0	1.2E-4	3.1E-2
1983	1.2E-4	6.2E-4	0.0E+0	5.2E-5	6.9E-6	1.0E-3	0.0E+0	4.5E-5	1.5E-2
1984	1.2E-4	4.2E-5	0.0E+0	8.0E-6	3.2E-6	1.4E-4	0.0E+0	5.5E-5	6.2E-3
1985	1.2E-4	9.0E-4	0.0E+0	2.3E-6	4.5E-7	1.0E-3	0.0E+0	6.6E-5	9.9E-2
1986	1.2E-4	3.8E-5	0.0E+0	5.5E-7	4.2E-8	1.0E-3	0.0E+0	6.9E-6	1.9E-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	1.2E-4	5.9E-6	0.0E+0	4.8E-7	7.3E-8	6.6E-3	0.0E+0	1.2E-5	2.2E-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	4.8E-5	0.0E+0	7.7E-8	1.3E-8	0.0E+0	0.0E+0	5.4E-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-2	0.0E+0	3.6E-4	3.6E-6	0.0E+0	0.0E+0	3.5E-2	0.0E+0
2001	0.0E+0	1.0E-2	0.0E+0	2.4E-6	2.9E-4	0.0E+0	0.0E+0	1.2E-3	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

Table 4-8. Intake (Bq yr^{-1}) by Year for TRA 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	1.2E+1	1.9E+0	0.0E+0	4.6E-3	6.8E-4	5.3E+1	0.0E+0	3.3E+0	5.0E+0
1967	3.0E-1	7.9E-1	0.0E+0	5.2E-4	7.6E-5	7.4E+0	0.0E+0	9.0E-1	2.8E+0
1968	2.1E+1	1.4E+0	0.0E+0	1.0E-2	1.5E-3	3.2E+0	0.0E+0	5.2E+0	2.7E+0
1969	1.2E+0	2.1E+0	0.0E+0	2.1E-3	3.2E-4	1.6E+0	0.0E+0	1.5E+0	2.3E+0
1970	2.8E+0	1.1E-4	0.0E+0	3.0E-3	4.5E-4	1.3E+0	0.0E+0	1.1E+0	2.3E+0
1971	1.1E+1	3.0E+0	0.0E+0	9.0E-3	1.3E-3	1.4E+1	0.0E+0	4.8E+0	2.0E+0
1972	1.2E+0	1.2E+0	0.0E+0	2.8E-3	4.2E-4	2.1E+0	0.0E+0	1.2E+0	7.2E-1
1973	5.7E-2	3.2E-5	0.0E+0	5.6E-4	8.1E-5	2.7E-1	0.0E+0	1.5E-1	5.1E-2
1974	2.4E-2	3.8E-3	0.0E+0	3.3E-4	2.9E-5	1.3E-1	0.0E+0	1.1E-1	3.7E-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	2.2E-4	1.4E-4	0.0E+0	3.5E-5	1.5E-5	3.5E-3	0.0E+0	1.7E-3	1.3E-1
1977	2.9E-4	2.0E-4	0.0E+0	1.1E-4	4.8E-5	1.5E-2	0.0E+0	7.2E-3	6.1E-1
1978	1.6E-3	8.7E-3	0.0E+0	3.2E-4	3.4E-5	2.4E-2	0.0E+0	8.3E-3	1.5E+0
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	4.1E-4	3.0E-3	0.0E+0	7.6E-6	1.5E-6	3.3E-3	0.0E+0	2.2E-4	3.3E-1
1986	4.1E-4	1.3E-4	0.0E+0	1.8E-6	1.4E-7	3.3E-3	0.0E+0	2.3E-5	6.2E-2
1987	1.2E-3	1.8E-4	0.0E+0	5.9E-6	9.0E-7	1.3E-4	0.0E+0	9.7E-5	3.3E+0
1988	1.2E-3	5.9E-5	0.0E+0	4.8E-6	7.3E-7	6.6E-2	0.0E+0	1.2E-4	2.2E+0
1989	1.2E-3	4.2E-5	0.0E+0	2.5E-8	3.5E-9	6.9E-4	0.0E+0	3.2E-5	2.9E-1
1990	4.1E-4	1.3E-4	0.0E+0	3.1E-9	3.1E-9	1.4E-4	0.0E+0	6.6E-7	1.0E-1
1991	4.1E-4	5.2E-5	0.0E+0	3.3E-10	3.3E-10	1.7E-4	0.0E+0	3.2E-4	6.0E-2
1992	4.1E-4	6.0E-5	0.0E+0	5.7E-7	5.7E-7	4.6E-5	0.0E+0	2.8E-5	1.0E-1
1993	0.0E+0	3.8E-5	0.0E+0	0.0E+0	8.7E-10	3.5E-4	0.0E+0	3.8E-4	0.0E+0
1994	0.0E+0	4.4E-5	0.0E+0	0.0E+0	6.6E-8	0.0E+0	0.0E+0	1.1E-4	0.0E+0
1995	0.0E+0	4.8E-5	0.0E+0	7.7E-8	1.3E-8	0.0E+0	0.0E+0	5.4E-6	0.0E+0
1996	0.0E+0	6.5E-5	0.0E+0	5.1E-7	1.0E-8	0.0E+0	0.0E+0	2.5E-6	0.0E+0
1997	0.0E+0	0.0E+0	0.0E+0	4.1E-7	1.3E-7	0.0E+0	0.0E+0	5.7E-5	0.0E+0
1998	0.0E+0	5.4E-5	0.0E+0	4.0E-7	4.3E-8	0.0E+0	0.0E+0	2.5E-5	0.0E+0
1999	0.0E+0	7.2E-5	0.0E+0	1.8E-7	1.7E-8	0.0E+0	0.0E+0	1.0E-5	0.0E+0
2000	0.0E+0	6.4E-3	0.0E+0	1.2E-4	1.2E-6	0.0E+0	0.0E+0	1.2E-2	0.0E+0
2001	0.0E+0	3.3E-3	0.0E+0	8.0E-7	9.5E-5	0.0E+0	0.0E+0	3.9E-4	0.0E+0
2002	0.0E+0	1.6E-3	0.0E+0	6.7E-5	1.8E-4	0.0E+0	0.0E+0	3.8E-2	0.0E+0

Table 4-9. Comparison of Calculated Facility Intakes with Intakes from Environmental Monitoring Results .

Year	*Activity Type*	*Average Annual Concentration*	*Annual Inhaled Quan. (Bq)*	*Table 4 Inhaled Quan. (Bq)*
1963	β-γ	1.7E-11 μCi/cc	1510	1310[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	^{90}Sr	3.4 fCi/M^3	0.3	0.15[b]
	^{95}Nb	1.0 fCi/M^3	0.09	---
	^{137}Cs	7-17 fCi/M^3	0.6-1.5	---
	^{144}Ce	4-8 fCi/M^3	0.36-0.71	0.057[b]
EFS	^{90}Sr	5.9 fCi/M^3	0.52	0.15[a]
	^{95}Nb	0.9-2.4 fCi/M^3	0.08-0.2	---
	^{106}Ru	6-9.8 fCi/M^3	0.53-0.87	0.27[a]
	^{134}Cs	0.8-1.6 fCi/M^3	0.07-0.14	---
	^{137}Cs	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	^{85}Kr @ CFA	3.7E-11 μCi/ml	3290	890[e]
1988	^{85}Kr @ CFA	1.1E-10 μCi/ml	9770	14,000[e]
1990	^{85}Kr @ CFA	2.7E-11 μCi/ml	2400	690[e]

a – values taken from Table 4-3 for CFA.
b – Values taken from 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 3E-14 and 6E-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 were derived from tables in an earlier version of the TBD report that contained concentrations of all INEEL released radionuclides.

Table 4-10: Intakes (Bq/event) for Criticalities that occurred at the INEEL.

Date	10/16/1959	10/16/1959	1/25/1961	1/25/1961	10/17/1978	10/17/1978	11/18/1958
Event	Criticality	Criticality	Criticality	Criticality	Criticality	Criticality	IET 13
Exposure Location	South ICPP	CFA Area	South ICPP	CFA Area	South ICPP	CFA Area	LPTF/STPF
Rb-89	5.2E+5	8.8E+4	1.5E+1	1.0E+0	2.2E+3	3.7E+4	8.1E+2
Sr-91	5.4E+4	9.2E+3	1.7E+2	1.1E+1	1.7E+1	7.0E+2	4.1E+2
Sr-92	6.5E+4	1.1E+4	1.5E+2	9.7E+0	5.6E+0	2.1E+2	1.9E+2
Y-92	7.5E+3	1.3E+3	7.3E+1	4.9E+0	9.1E-2	2.6E+1	5.6E+1
Y-93	5.9E+3	1.0E+3	1.9E+1	1.3E+0	1.3E-1	1.5E+1	2.0E+1
Te-133					1.1E+3	8.0E+3	
I-131(elem.)	2.9E+2	5.0E+1	1.9E+0	1.3E-1	3.9E-2	1.7E+0	3.4E+0
I-133	6.2E+3	1.1E+3	3.8E+1	2.5E+0	4.4E+0	5.8E+2	7.1E+1
I-134	4.6E+4	7.8E+3	7.2E+1	4.8E+0	1.1E+2	3.4E+3	2.0E+2
I-135	2.0E+4	3.4E+3	1.1E+2	7.2E+0	5.5E+1	2.2E+3	4.6E+2
Cs-138	7.7E+5	1.3E+5			5.2E+2	5.6E+4	5.0E+3
Ba-139	4.0E+5	6.8E+4	6.6E+2	4.4E+1	1.4E+2	1.8E+4	3.4E+3
La-141	2.8E+4	4.7E+3	1.0E+2	6.8E+0	4.6E-1	9.0E+1	2.0E+2
La-142	2.3E+4	3.9E+3	4.3E+1	2.8E+0	4.8E-1	5.7E+1	9.3E+1
U-234							7.1E-2
Dose Fract	93.9%	92.5%	94.5%	94.5%	98.3%	97.1%	94.4%

Table 4-11: Intakes (Bq/event) for Special Tests at the INEEL .

Date	Event	Exposure Location	Sr-89	Sr-90	Y-91	Zr-95	Ru-103	Ru-106	I-131 (elem.)	Ce-144	Pr-143
3/20/1957	FEBT A (14:19)	TAN	3.9E-2	6.1E-4	4.7E-2	5.0E-2	2.5E-2	1.1E-3	2.0E-2	1.8E-2	1.3E-2
3/20/1957	FEBT B (18:47)	TAN	1.3E-2	3.8E-3	2.4E-2	3.2E-2	3.8E-3	4.5E-3	9.0E-9	6.5E-2	2.6E-6
10/29/58 10/30/58	FECF Filter Break	South ICPP	4.2E-2	4.2E-2	9.1E-2	1.3E-1	8.3E-3	4.3E-2	2.7E-13	5.9E-1	1.8E-7
10/29/58 10/30/58	FECF Filter Break	CFA Area	1.8E+0	1.8E+0	3.9E+0	5.8E+0	3.6E-1	1.9E+0	1.2E-11	2.6E+1	7.6E-6

Table 4-12: Intakes (Bq/event) for Initial Engine Tests at the 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	TRA/ICPP	7.6E+0	2.3E-2	2.7E+0	1.7E+1	2.4E+1	5.5E+1	4.8E-6
4/24 - 5/19/59	IET 14	CFA	6.4E+0	2.0E-2	2.3E+0	1.4E+1	2.0E+1	4.6E+1	4.0E-6
6/16-6/24/59	IET 15(B)	TRA/ICPP	4.3E-1	1.1E-3	5.9E-1	2.6E+0	4.2E+0	3.0E+0	5.1E-5
6/16-6/24/59	IET 15(B)	CFA	3.6E-1	9.3E-4	5.0E-1	2.2E+0	3.6E+0	2.5E+0	4.3E-5
7/28 - 10/9/59	IET 16	SPERT	2.7E-2	3.6E-4	6.4E-4	1.3E-2	2.8E-2	9.7E-1	1.2E-7
7/28 - 10/9/59	IET 16	S. gate	2.2E-2	2.8E-4	5.1E-4	1.1E-2	2.2E-2	7.7E-1	9.7E-8
10/12-12/12/59	IET 17(B)	TRA/ICPP	2.9E-3	8.2E-3	7.5E-1	3.2E+0	3.3E+0	8.2E-1	7.2E-6
10/12-12/12/59	IET 17(B)	CFA	2.4E-3	6.9E-3	6.3E-1	2.7E+0	2.7E+0	6.9E-1	6.1E-6
1/26-2/7/60	IET 18	SPERT	1.2E-2	2.6E-3	8.3E+0	4.7E+1	3.1E+1	4.9E-1	8.1E-6
1/26-2/7/60	IET 18	S. gate	9.4E-3	2.0E-3	6.6E+0	3.7E+1	2.5E+1	3.9E-1	6.4E-6
2/17 - 2/29/60	IET 19(A)	TRA/ICPP	1.7E-1	8.9E-3	1.3E+0	8.8E+0	1.3E+1	6.4E+0	5.5E-7
2/17 - 2/29/60	IET 19(A)	CFA	1.4E-1	7.5E-3	1.1E+0	7.4E+0	1.1E+1	5.4E+0	4.6E-7
11/22-11/30/60	IET 25(A)	TRA/ICPP	9.3E-4	6.1E-4	4.7E-1	5.2E+0	6.7E+0	2.0E-1	2.0E-6
11/22-11/30/60	IET 25(A)	CFA	7.8E-4	5.2E-4	3.9E-1	4.3E+0	5.6E+0	1.7E-1	1.7E-6
12/1-12/15/60	IET 25(B)	SPERT	1.2E-3	8.0E-4	1.4E+0	8.7E+0	8.3E+0	1.3E-1	3.2E-6
12/1-12/15/60	IET 25(B)	S. gate	9.5E-4	6.4E-4	1.1E+0	6.9E+0	6.6E+0	1.1E-1	2.5E-6
12/23-12/28/60	IET 26(A)	TRA/ICPP	2.2E+0	1.5E-2	2.4E+0	9.5E+0	1.4E+1	2.0E+1	3.3E-5
12/23-12/28/60	IET 26(A)	CFA	1.9E+0	1.2E-2	2.0E+0	8.0E+0	1.1E+1	1.7E+1	2.8E-5

Table 4-13. INEEL facility fence direct gamma values (TLD – Background) (mR).

Year	ARA I & II	SPERT	TAN-TSF	TAN-LOFT	TAN-LPT	CFA	TRA	ICPP	RWMC	EBR-II	TREAT	Backgrnd
1952-72	226	12	42	10	12	52	438	446	32	36	18	100-150
1973	86	21	41	17	12	53	306	405	32	37	19	121
1974	162	48	8	7	0	59	320	627	370	35	17	123
1975	114	16	29	11	7	17	195	357	265	32	8	118
1976	66	27	22	15	12	20	140	311	155	56	50	113
1977	41	5	0	4	4	1	137	318	189	22	0	132
1978	52	12	2	9	4	7	143	251	106	56	2	129
1979	63	18	10	17	7	14	159	236	65	59	5	113
1980	65	17	8	19	13	18	251	203	57	51	12	119
1981	63	18	8	17	10	14	231	255	42	28	9	118
1982	50	26	6	12	6	10	163	124	42	20	12	117
1983	78	17	23	26	19	18	174	141	50	24	10	115
1984	80	19	11	19	12	15	205	181	48	31	13	124
1985	80	19	11	19	12	15	205	181	48	31	13	124
1986	80	19	11	19	12	15	205	181	48	31	13	124
1987	80	19	11	19	12	15	205	181	48	31	13	124
1988	80	19	11	19	12	15	205	181	48	31	13	124
1989	80	19	11	19	12	15	205	181	48	31	13	124
1990	80	10	10	11	9	11	28	39	27	19	13	124
1991	80	10	10	11	9	11	28	39	27	19	13	124
1992	80	10	10	11	9	11	28	39	27	19	13	124
1993	77	19	18	23	15	15	48	37	24	28	16	111
1994	69	0	0	0	0	0	24	28	25	15	3	130
1995	91	6	4	8	2	11	31	43	42	17	7	116
1996	52	4	14	0	0	13	28	49	40	22	21	129
1997	46	10	3	8	0	9	29	44	17	16	16	128
1998	62	8	0	5	0	12	25	31	20	0	11	131
1999	49	13	0	0	0	5	10	38	22	13	13	122
2000	28	16	7	16	8	19	40	55	61	25	26	129
2001	31	3	0	0	0	0	27	32	25	0	3	140
2002	41	11	0	0	0	9	34	54	33	18	39	120

REFERENCES

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

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

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

Brodsky, A., and G. V. Beard (eds.), 1960, *A Compendium of Information for Use in Controlling Radiation Emergencies, Including Lecture Notes from a Training Session in Idaho Falls, ID, February 12-14, 1958,* ID-8206, U.S. Department of Energy, Idaho Operations Office, Idaho Falls, Idaho.

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, T*he 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. (ed.), 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. F. Kunze, D. A. McKenzie, C. B. Cannon, C. Davidson, and A. F. Clark, 1960, *Preliminary Data Report IET Test Series No. 18 Phase II Testing of the D102A2 Powerplant*, DC 60-6-735, June 17.

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.

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. (ed.), 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.

Showalter, D. E. (ed.), 1959, *Preliminary Data Report IET 16 D102A2(A)*, XDCL 59-11-715.

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

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

Stoller (S. M. Stoller Corp.) 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., A. P. Richter, 1966, *Climatography of the National Reactor Testing Station,* IDO-12048.