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PUBLICATION RECORD

EFFECTIVE DATE	REVISION NUMBER	DESCRIPTION
03/30/2004	00	New Technical Basis Document for the Idaho National Engineering and Environmental Laboratory – Occupational Environmental Dose. First approved issue. Initiated by Norman D. Rohrig.
08/17/2007	01	Approved Revision 01 initiated to add additional information on how TLD results were chosen. Added Purpose, Scope, and Acronyms sections. Added references and SRDB numbers. Changed Idaho National Engineering and Environmental Laboratory to Idaho National Laboratory. Incorporates formal internal and NIOSH review comments. Adds Attributions and Annotations section. Incorporates formal internal and NIOSH review of the Attributions and Annotations section. Constitutes a total rewrite of document. Training Required: As determined by the Task Manager. Initiated by Jo Ann M. Jenkins.

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ACRONYMS AND ABBREVIATIONS

AEC	U.S. Atomic Energy Commission
AMAD	activity median aerodynamic diameter
ANL-W	Argonne National Laboratory–West
ARA	Army (later Auxiliary) Reactor Area
BORAX	Boiling-Water Reactor Experiment
Bq	becquerel
CFA	Central Facilities Area
Ci	curie
cm	centimeter
CPP	Chemical Processing Plant
d	day
DOE	U.S. Department of Energy
EBR	Experimental Breeder Reactor
EEOICPA	Energy Employee Occupational Illness Compensation Program Act of 2000
EFS	Experimental Field Station
EMR	Environmental Monitoring Report
EOCR	Experimental Organic-Cooled Reactor
F	fast (solubility rate)
fCi	femtocurie
FEBT	Fuel Element Burn Test
FECF	Fuel Element Cutting Facility
FPFRT	Fission Product Field Release Test
ft	foot
GE-ANP	General Electric–Aircraft Nuclear Propulsion (program)
H&S	Health and Safety (division)
hr	hour
HTRE	Heat Transfer Reactor Experiment
ICPP	Idaho Chemical Processing Plant
IDO	Idaho Operations Office
IET	Initial Engine Test
in.	inch
INEEL	Idaho National Engineering and Environmental Laboratory
INEL	Idaho National Engineering Laboratory
INL	Idaho National Laboratory
INTEC	Idaho Nuclear Technology and Engineering Center
km	kilometer
kW	kilowatt
LOFT	Loss-of-Fluid Test
LPTF	Low Power Test Facility

m	meter
M	moderate (solubility rate)
mi	mile
min	minute
mL	milliliter
mo	month
mR	milliroentgen
mrem	millirem
MW	megawatt
NCRP	National Council on Radiation Protection and Measurements
NIOSH	National Institute for Occupational Safety and Health
NOAA	National Oceanic and Atmospheric Administration
NRTS	National Reactor Testing Station
NSA	North Storage Area
ORAU	Oak Ridge Associated Universities
PBF	Power Burst Facility
pCi	picocurie
POC	probability of causation
RSAC	Radiological Safety Analysis Computer (program)
RWMC	Radioactive Waste Management Complex
S	slow (solubility rate)
s	second
SL-1	Stationary Low-Power Reactor No. 1
SPERT	Special Power Excursion Reactor Test
STPF	Shield Test Pool Facility
TAN	Test Area North
TBD	technical basis document
TLD	thermoluminescent dosimeter
TRA	Test Reactor Area
TREAT	Transient Reactor and Experiment Test (facility)
TSF	Technical Support Facility
U.S.C.	United States Code
WWP	Warm Waste Pond
yr	year
β	beta
γ	gamma
µCi	microcurie
°F	degrees Fahrenheit

§ section or sections

4.1 INTRODUCTION

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

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

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

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

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

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

4.1.1 Purpose

This TBD provides the basis for the environmental doses at the Idaho National Laboratory (INL) and its predecessor organizations [the National Reactor Testing Station (NRTS), the Idaho National Engineering Laboratory (INEL), and the Idaho National Engineering and Environmental Laboratory (INEEL)]. This information can be used in dose reconstructions for the EEOICPA to supplement information in an individual's dose record.

This TBD addresses radioactive material releases from areas or facilities at the INL that could affect employees at another facility. The releases discussed here have been divided into (1) normal operational releases and (2) episodic releases that generally have been of short duration. These radioactive material releases potentially represent unrecorded or missed doses, either as direct gamma or beta-gamma from immersion in the radioactive gaseous cloud for individuals who did not have personal dosimetry to record the dose or as internal doses from inhalation of the materials.

This TBD also addresses direct gamma doses from facility operations. In general, these doses, if not controlled by management, increase with time and create a *facility background dose*. At INL, 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 doses, or facility *fence-line* doses, as they are sometimes called, are a nebulous indication of a dose that individuals could receive if they worked at the facility. INL facility fence-line doses (minus background) are presented for 11 locations.

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

4.1.2 Scope

As outlined and discussed in the latest revision of Section 2 of this Site Profile (ORAUT 2005), the INL site was chosen by the U.S. Atomic Energy Commission (AEC) as an isolated location for testing various reactor concepts. INL 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 INL sits above the large Snake River Aquifer that eventually surfaces and enters the Snake River in the Hagerman Valley area, the annual flow rate of the water in the aquifer is measured at 5 to 15 ft/d (ERDA 1977a).

During the 50-yr history of the site about 50 different reactor concepts have been designed, built, and operated at INL. All of these reactors have been prototype, low-power critical, or test reactors; no weapons production or commercial power reactors have been operated at INL. 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 Test Reactor, Engineering Test Reactor, and Advanced Test Reactor), all at the Test Reactor Area (TRA) near the middle southern end of the site, and (3) the Experimental Breeder Reactor No. 2 (EBR-II), at Argonne National Laboratory–West (ANL-W) at the southeastern corner of the site (Osloond 1965; Osloond and Newcomb 1969).

Another historically important airborne effluent producer is the Idaho Nuclear Technology and Engineering Center (INTEC), formerly known as the Idaho Chemical Processing Plant (ICPP). This facility, constructed in the early 1950s, began processing nuclear fuel in February 1953 and continued until 1992. Throughout its history, the Chem Plant, or CPP as it is commonly known, has reprocessed

fuel from test reactors at INL, zirconium-clad fuel reclaimed from various reactors, stainless-steel-clad fuel from EBR-II, and many AEC test reactors from around the world. Apart from the GE-ANP Program, which will be discussed below, INTEC airborne releases have historically been the most radiologically significant releases at INL (DOE 1991; Till et al. 2002). Through the years that INL Environmental Monitoring Reports (EMRs) have been published, INTEC airborne effluents have been attributed to creating the maximum INL boundary dose. Considering this fact, it should be suspected that INTEC airborne effluent would also be responsible for the maximum INL worker doses. Figure 4-1 shows the locations of INL facilities.

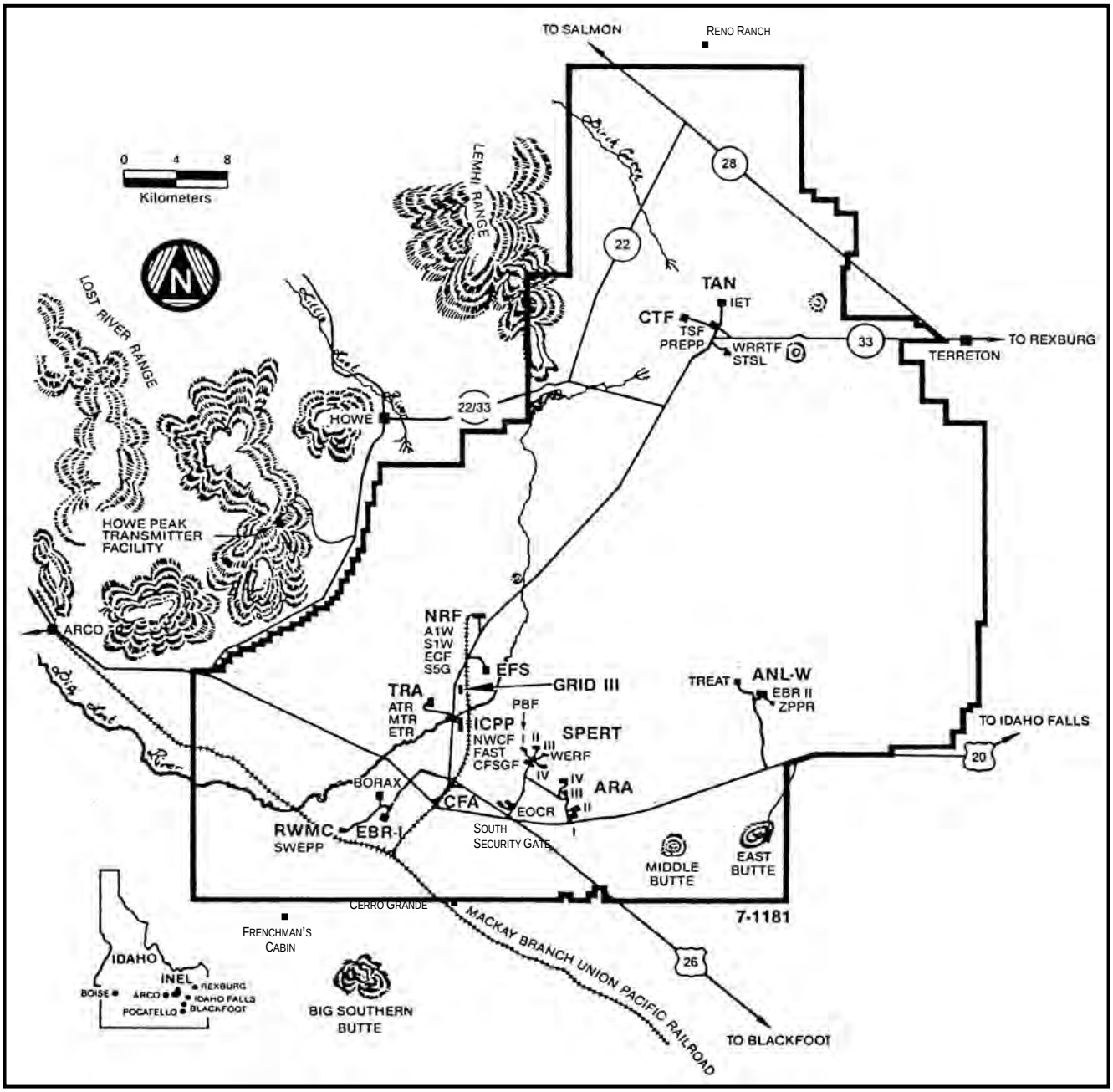

Figure 4-1. INL site map (Hoff, Chew, and Rope 1987, p. 46).

From the beginning of operations at the INL site, facility locations were chosen to limit the potential for operational releases at one facility to affect another facility. Because the site encompasses 890 mi^2, there was ample room to place facilities with this principle in mind. Because the site has an average elevation of 5,000 ft and 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-yr history of the site has demonstrated that the large expanse of INL and this meteorological characteristic have been satisfactorily effective in maintaining the operational isolation of each facility [1].

Beginning with the GE-ANP Program, which began in the early 1950s, the 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 with a rather extensive network of meteorological monitoring stations (Yanskey, Markee, and Richter 1966). The plumes from all intentional planned releases [Controlled Environmental Radioiodine (later Release) Test (CERT), Fission Product Field Release Test (FPFRT), Fuel Element Burn Tests (FEBTs) 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 [2]. In general, these tests were performed in support of a specific program (i.e., the FPFRT and FEBTs A and B were conducted to support the GE-ANP Program).

All of the airborne releases at INL that have occurred since the beginning of operations were reviewed and analyzed as a result of a request from the DOE Idaho Operations Office (IDO). This request was to evaluate the radiological impact to INL 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 INL. 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). All releases considered for that report are the basis for the releases considered in this TBD. In addition, all releases documented in the evaluation, operational and episodic, have been independently reviewed and found, with minor modifications, to be substantially appropriate. The review, conducted by the Radiological Assessment Corporation at the request of the Centers for Disease Control and Prevention and the State of Idaho, also evaluated the methodology by which the RSAC-4 program performs dose calculations against the National Council on Radiological Protection and Measurements (NCRP) methodology (Till et al. 2002). It stated:

> *As a final point, Tables 7, 8, 9a, 9b, 10a, and 10b, and Figures 18 and 19 confirm that the NCRP method was suitable for these ranking purposes when the results are compared with those using the RSAC code. In all cases, the RSAC code confirmed the results obtained using the NCRP methodology.* (Till et al. 2002, p. 57)

Version 6 of the RSAC code (Wenzel and Schrader 2001) was used extensively for this 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 facilities and programs at the INL. A few comments on the development of those facilities will clarify some of the questions that might arise when viewing the INL 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 the site,

and they were operated under the AEC Chicago Operations Office by the University of Chicago as ANL-W. These essentially low-power reactors produced very little radioactive airborne effluent (Osloond 1965; Osloond and Newcomb 1969). As the EBR-I and BORAX programs were completed, ANL-W relocated to the eastern section of the site near the EBR-II, the Transient Reactor and

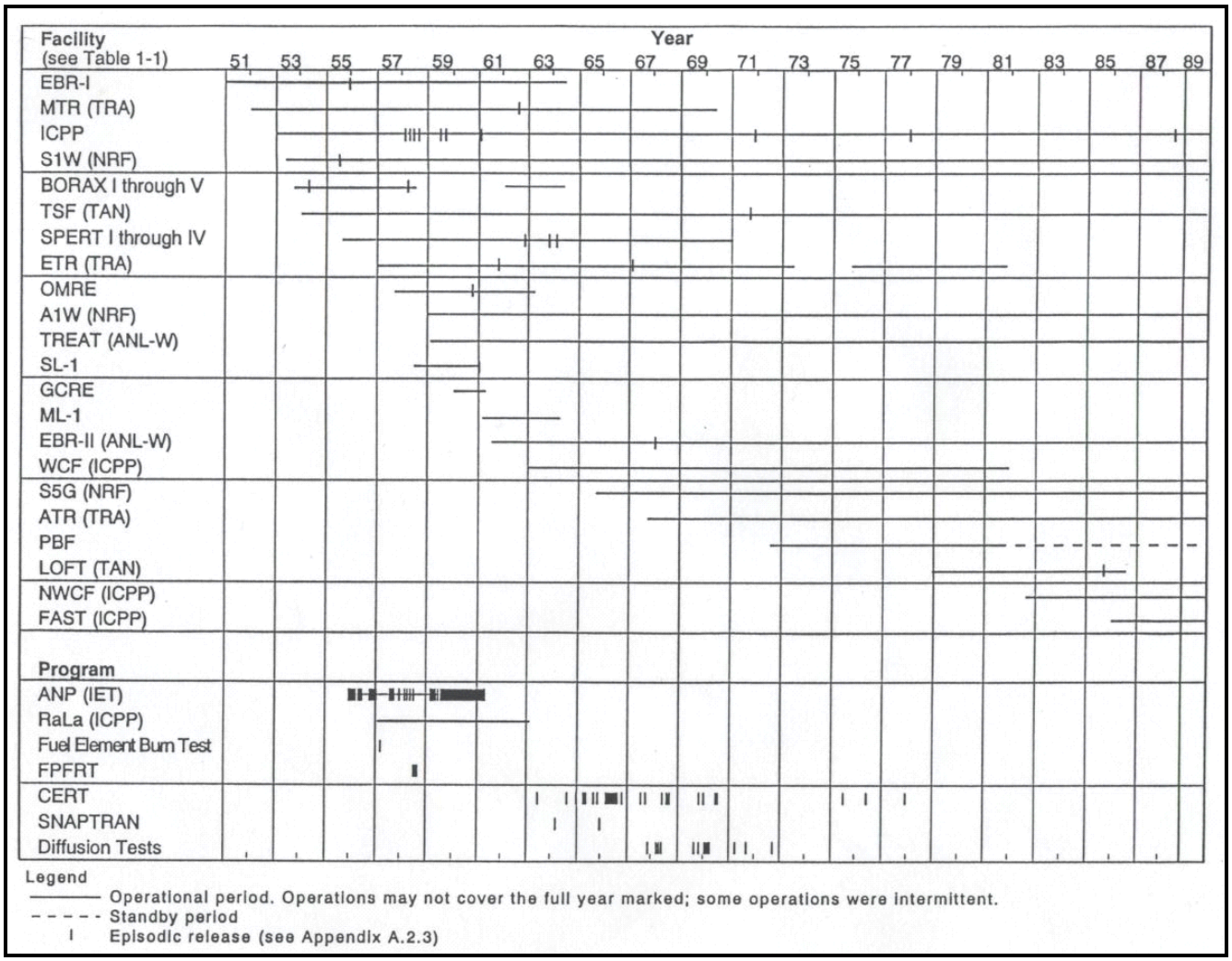

Figure 4-2. Chronology of facilities and programs at the INL site (DOE 1991).

Experiment Test (TREAT) Facility, the Zero Power Plutonium (later Physics) Reactor, etc. The EBR-I location is now a National Historic Landmark. The Stationary Low-Power Reactor (SL-1), the Mobile Low-Power Reactor No. 1, and the Gas-Cooled Reactor Experiment, which were operated for the U.S. Army, were at the Army (later Auxiliary) Reactor Area (ARA). The NRTS Burial Ground became the Radioactive Waste Management Complex (RWMC). The short-lived Organic-Moderated Reactor Experiment and Experimental Organic-Cooled Reactor (EOCR) were at the EOCR location.

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

All inhaled quantities and concentrations referred to in this document apply to individuals stationed at the site. DOE and INL contractor employees in Idaho Falls, Idaho, in IDO contractor facilities (Willow Creek Building, the INEL Research Center, the Computer Science Center, etc.) are not affected by site activities and, thus, are not subject to inhaled quantities and concentrations [3].

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 at INTEC and HTRE No. 3 during Initial Engine Test (IET) No. 13, FEBTs, etc. As stated above, operational releases from INTEC and TRA have been the predominant and radiologically significant releases at INL during the history of the site. For more discussion of these releases and their relationship to other, less significant releases, see Peterson (2004), DOE (1991), and Till et al. (2002).

For worker dose reconstruction, the analyst should use default values for the calculation under consideration. When solubility is of concern, use an insoluble oxide form for solids for analysis, with type S and M materials being the predominant form [4]. Without more definitive information on the type of material, use the material that maximizes the dose for a particular situation. When iodines are of concern, consider them to be type F [5].

4.2.1 Operational Releases

To determine onsite concentrations of radionuclides from operational releases at INL facility locations, the same methodology used to determine offsite concentrations for annual EMRs is used. The release for each year of operation is exactly the same as that documented in 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 INL facility were picked from the annual average mesoscale dispersion isopleths of ground-level air concentrations published in the EMRs for INL, as described in DOE (1991). As described in Appendix B of that document, dispersion isopleths are available for the years beginning in 1973, with the exception of 1978 when the telemetry system was upgraded. For the years before 1973, the 9-yr average of mesoscale dispersion isopleths of ground-level air concentrations (DOE 1991) shown in Figure 4-3 was used.[2] For 1993 to 2005, annual average mesoscale isopleths from the annual environmental reports (Mitchell 1994; Mitchell, Peterson, and Hoff 1995; Mitchell et al. 1996; Mitchell et al. 1997; Evans et al. 1998; Saffel et al. 2000; Stoller 2002a,b,c, 2004, 2005,2007) were used to calculate the facility annual concentration.

Of the many facilities on INL, eight facility areas have been chosen for analysis: TAN, INTEC, TRA, RWMC, CFA, SPERT, ARA, and ANL-W. TAN includes the IET, Core Test Facility, Technical Support Facility (TSF), Loss-of-Fluid Test (LOFT) Facility, Specific Manufacturing Capability, Water Reactor Research Test Facility, Process Experimental Pilot Plant, and Low Power Test Facility (LPTF), depending on the year of operation (ORAUT 2005). The SPERT area includes the Waste Experimental Reduction Facility and the Power Burst Facility (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. An isopleth chosen for a given year for a particular facility, such as SPERT, is assumed to apply to PBF, SPERT-I, SPERT-II, etc. If a facility was between two isopleths,

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

the higher-valued isopleth was chosen. Yearly isopleth values for each of the eight facilities have been extracted from the annual EMRs and converted from the normalized annual concentration[2] (hr^2/m^3) to concentrations (becquerels per cubic meter) and multiplied by 2.4×10^3 m^3/yr (the amount of air breathed occupationally per year) to produce activity inhaled per year (becquerels) for an occupational individual. These are listed in Tables 4-1 through 4-8 for each of the facility areas.

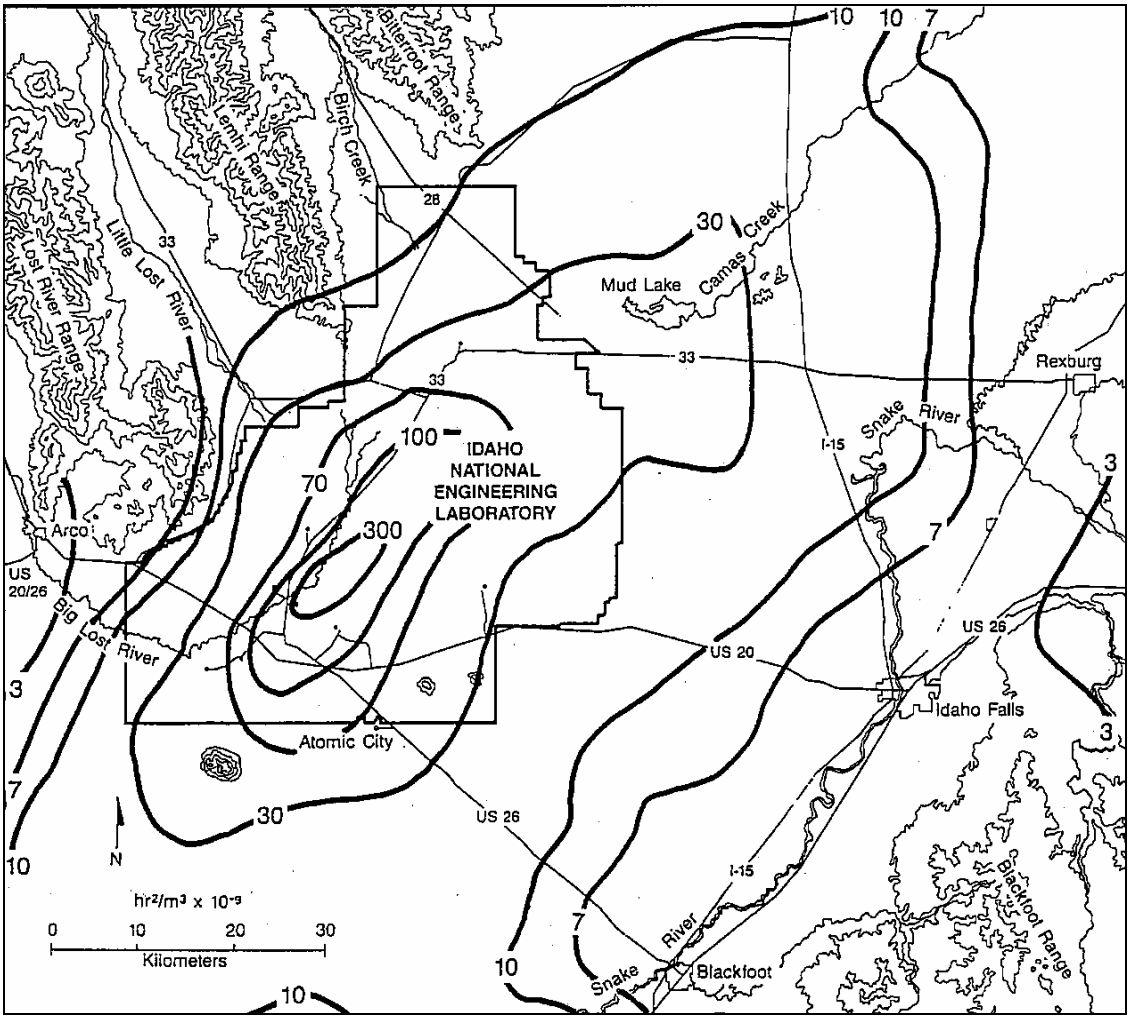

Figure 4-3. Nine-year (1974 to 1983 excluding 1978) average mesoscale dispersion isopleths of air concentration at ground level ($hr^2/m^3 \times 10^{-9}$), normalized to unit release rate (DOE 1991).

The annual inhaled quantities (becquerels per year) listed in Tables 4-1 through 4-8 are based on known and published annual airborne emissions. The following discussion provides information found in site documentation about facility environmental sampling and monitoring and provides data that can be compared with these calculations.

Environmental air sampling at the facility areas has been performed at least since the mid-1950s where airborne effluents were known or suspected to exist [6]. The early IDO Health and Safety (H&S) Division annual reports (AEC 1960 -1970b) 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 were not performed in a set facility location or for a standard duration. Some facility data are presented in the 1963 Annual Progress Report of the H&S Division

(Dodd 1964) and a routine facility environmental monitoring program was developed between 1963 and 1970. In 1968 and 1969, formal annual EMRs report alpha, beta, and 131I concentrations that can be correlated with the values in Tables 4-1 through 4-8. The 1970 EMR (AEC 1971, p. 5) discusses 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 (AEC 1971, 1972, 1973, 1974; ERDA 1975, 1976, 1977b; DOE 1978a,b, 1979a, 1980a, 1981a, 1982a,

Table 4-1. Intake (Bq/yr) by year for ANL-W, 1952 to 2005 [7].

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
2003	9.4E-8	1.2E-2	2.4E-2	1.0E-5	0.0E+0	0.0E+0	0.0E+0	0.0E+0	0.0E+0
2004	1.8E-5	1.2E-4	4.9E-4	4.0E-5	5.6E-5	0.0E+0	5.9E-6	1.1E-3	0.0E+0
2005	8.8E-7	2.3E-2	8.4E-5	4.5E-5	2.6E-4	4.8E-8	3.8E-4	1.3E-2	0.0E+0

Table 4-2. Intake (Bq/yr) by year for ARA, 1952 to 2005 [7].

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
2003	5.7E-8	7.2E-3	1.4E-2	6.3E-6	0.0E+0	0.0E+0	0.0E+0	0.0E+0	0.0E+0
2004	6.9E-6	4.4E-5	1.8E-4	1.5E-5	2.1E-5	0.0E+0	2.2E-6	4.0E-4	0.0E+0
2005	8.8E-7	2.3E-2	8.4E-5	4.5E-5	2.6E-4	4.8E-8	3.8E-4	1.3E-2	0.0E+0

Table 4-3. Intake (Bq/yr) by year for CFA, 1952 to 2005 [7].

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
2003	3.8E-7	4.8E-2	9.5E-2	4.2E-5	0.0E+0	0.0E+0	0.0E+0	0.0E+0	0.0E+0
2004	2.8E-5	1.8E-4	7.3E-4	6.0E-5	8.4E-5	0.0E+0	8.9E-6	1.6E-3	0.0E+0
2005	1.9E-6	4.9E-2	1.8E-4	9.6E-5	5.6E-4	1.0E-7	8.1E-4	2.9E-2	0.0E+0

Table 4-4. Intake (Bq/yr) by year for INTEC, 1952 to 2005 [7].

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
2003	5.7E-7	7.2E-2	1.4E-1	6.3E-5	0.0E+0	0.0E+0	0.0E+0	0.0E+0	0.0E+0
2004	3.4E-5	2.2E-4	9.1E-4	7.6E-5	1.0E-4	0.0E+0	1.1E-5	2.0E-3	0.0E+0
2005	3.8E-6	9.8E-2	3.6E-4	1.9E-4	1.1E-3	2.1E-7	1.6E-3	5.8E-2	0.0E+0

Table 4-5. Intake (Bq/yr) by year for RWMC, 1952 to 2004 [7].

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
2003	1.9E-7	2.4E-2	4.7E-2	2.1E-5	0.0E+0	0.0E+0	0.0E+0	0.0E+0	0.0E+0
2004	2.5E-5	1.6E-4	6.7E-4	5.5E-5	7.7E-5	0.0E+0	8.1E-6	1.5E-3	0.0E+0
2005	8.8E-7	2.3E-2	8.4E-5	4.5E-5	2.6E-4	4.8E-8	3.8E-4	1.3E-2	0.0E+0

Table 4-6. Intake (Bq/yr) by year for SPERT, 1952 to 2005 [7].

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
2003	1.1E-7	1.4E-2	2.8E-2	1.3E-5	0.0E+0	0.0E+0	0.0E+0	0.0E+0	0.0E+0
2004	1.1E-5	7.4E-5	3.0E-4	2.5E-5	3.5E-5	0.0E+0	3.7E-6	6.7E-4	0.0E+0
2005	6.3E-7	1.6E-2	6.0E-5	3.2E-5	1.9E-4	3.5E-8	2.7E-4	9.6E-3	0.0E+0

Table 4-7. Intake (Bq/yr) by year for TAN, 1952 to 2005 [7].

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
2003	3.8E-8	4.8E-3	9.5E-3	4.2E-6	0.0E+0	0.0E+0	0.0E+0	0.0E+0	0.0E+0
2004	3.4E-6	2.2E-5	9.1E-5	7.6E-6	1.0E-5	0.0E+0	1.1E-6	2.0E-4	0.0E+0
2005	1.9E-7	4.9E-3	1.8E-5	9.6E-6	5.6E-5	1.0E-8	8.1E-5	2.9E-3	0.0E+0

Table 4-8. Intake (Bq/yr) by year for TRA, 1952 to 2005 [7].

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
2003	2.8E-7	3.6E-2	7.1E-2	3.1E-5	0.0E+0	0.0E+0	0.0E+0	0.0E+0	0.0E+0
2004	3.4E-5	2.2E-4	9.1E-4	7.6E-5	1.0E-4	0.0E+0	1.1E-5	2.0E-3	0.0E+0
2005	1.9E-6	4.9E-2	1.8E-4	9.6E-5	5.6E-4	1.0E-7	8.1E-4	2.9E-2	0.0E+0

1983a; Hoff, Chew, and Dickson 1984, 1985; Hoff, Chew, and Rope 1986, 1987; Chew and Mitchell 1988; Hoff, Mitchell, and Moore 1989; Hoff et al. 1990, 1991).

Table 4-9 lists results of the comparison. Because of the large variation in measurements made, the ratio of values calculated from the EMRs (column 5) to that derived from releases listed in Table 4-1 through 4-8 is not well-behaved. Three of the 13 values with uncertainties overlap. The geometric mean of the remainder of values is 0.36 with a geometric standard deviation of 5.3. Nevertheless, this comparison provides confidence in the results of Table 4-1. The value at the upper 84% confidence value 1.9 (0.36*5.3) is less than the default geometric standard deviation of 3 assumed for environmental results [8].

Table 4-9. Comparison of calculated facility intakes with intakes from environmental monitoring results.

Year	Activity type	Average annual Concentration	Reference for Col. 3	Annual inhaled quantity (Bq)	INL TBD table inhaled quantity (Bq)
1963	β-γ	17 pCi/m^3		1,510	1,310[a]
	Pu-239	0.6 fCi/m^3		0.05	0.014[a]
1968	α	2.2 fCi/m^3	AEC 1969a, p14	0.18	0.01[a]
	β	0.64 pCi/m^3	AEC 1969a, p14	56	337[a]
	I-131	<0.08 pCi/m^3	AEC 1969a, p14	<7.1	1.4[a]
1969	α	0.003 pCi/m^3	AEC 1970a, p14	0.27	2.4E-3[a]
	β	0.501 pCi/m^3	AEC 1970a, p14	44	118[a]
	I-131	<0.014 pCi/m^3	AEC 1970a, p14	<1.2	2.1[a]
1970	Gross β	0.6 pCi/m^3	AEC 1971, p 5	53	74[a]
	Max. gr. β at CFA	0.81 pCi/m^3	AEC 1971, p 5	72	74[a]
1973	Gross β	95 ± 42 fCi/m^3	AEC 1974, p 14	8.4 ± 3.7	0.8[a]
EBR-I	Sr-90	3.4 ±3.0 fCi/m^3	AEC 1974, p 16	0.3 ± 0.27	0.15[b]
	Nb-95	1.0-2.5 fCi/m^3	AEC 1974, p 16	0.09 – 0.22	---
	Cs-137	7-17 fCi/m^3	AEC 1974, p 16	0.6 - 1.5	---
	Ce-144	4-8 fCi/m^3	AEC 1974, p 16	0.36 - 0.71	0.057[b]
EFS	Sr-90	5.9±8.6 fCi/m^3	AEC 1974, p 16	0.52 ± 0.76	0.15[a]
	Nb-95	0.9-2.4 fCi/m^3	AEC 1974, p 16	0.08 - 0.21	---
	Ru-106	6-9.8 fCi/m^3	AEC 1974, p 16	0.53 - 0.87	0.27[a]
	Cs-134	0.8-1.6 fCi/m^3	AEC 1974, p 16	0.07 - 0.14	---
	Cs-137	17-27 fCi/m^3	AEC 1974, p 16	1.5 - 2.4	---
1976[c]	Gross β	30-60 fCi/m^3	ERDA 1977b	2.5 – 5.3	0.6-25[d]
1986	Kr-85 at CFA	37±15 pCi/m^3	Hoff et al. 1987	3,290 ± 1330	890[e]
1988	Kr-85 at CFA	108 ± 69 pCi/m^3	Hoff et al. 1989	9,770 ± 6130	14,000[e]
1990	Kr-85 at CFA	27.7 pCi/m^3	Hoff et al. 1991	2,400	690[e]

a. Values from INEEL TBD Table 4-3 for CFA.
b. Values from INEEL TBD Table 4-5 for RWMC since EBR-I is near RWMC.
c. Of 90 monthly values (January through September) for 10 facility areas, 89 values were between 3×10^{-14} and 6×10^{-14} µCi/mL.
d. Using the current tables with 11 radionuclides, the inhaled quantity is about 0.6 Bq; with the original tables with 44 radionuclides, the inhaled quantity is about 25 Bq.
e. Values derived from tables in Peterson (2004) that contained concentrations of all INEEL released radionuclides.

Figure 4-4 shows the variation of the INL environmental monitoring sampling results for the 9-yr period from 1978 through 1986. This figure also shows the close correlation of environmental sample results acquired at *distant communities* and those acquired at INL facilities as well as the effect of foreign nuclear tests and the Chernobyl reactor accident on INL environmental sampling results. As shown on this figure, the INL average concentration for the 9-yr period has not differed from the distant community concentrations by more than a factor of 2. Inspection of subsequent year EMRs shows the same is true for the years after 1986 (Hoff, Chew, and Rope 1987; Chew and Mitchell

1988; Hoff, Mitchell, and Moore 1989; Hoff et al. 1990, 1991, 1992, 1993; Mitchell 1994; Mitchell, Peterson, and Hoff 1995; Mitchell et al. 1996, 1997; Evans et al. 1998; Saffel et al. 2000; Stoller 2002a,b,c, 2003).

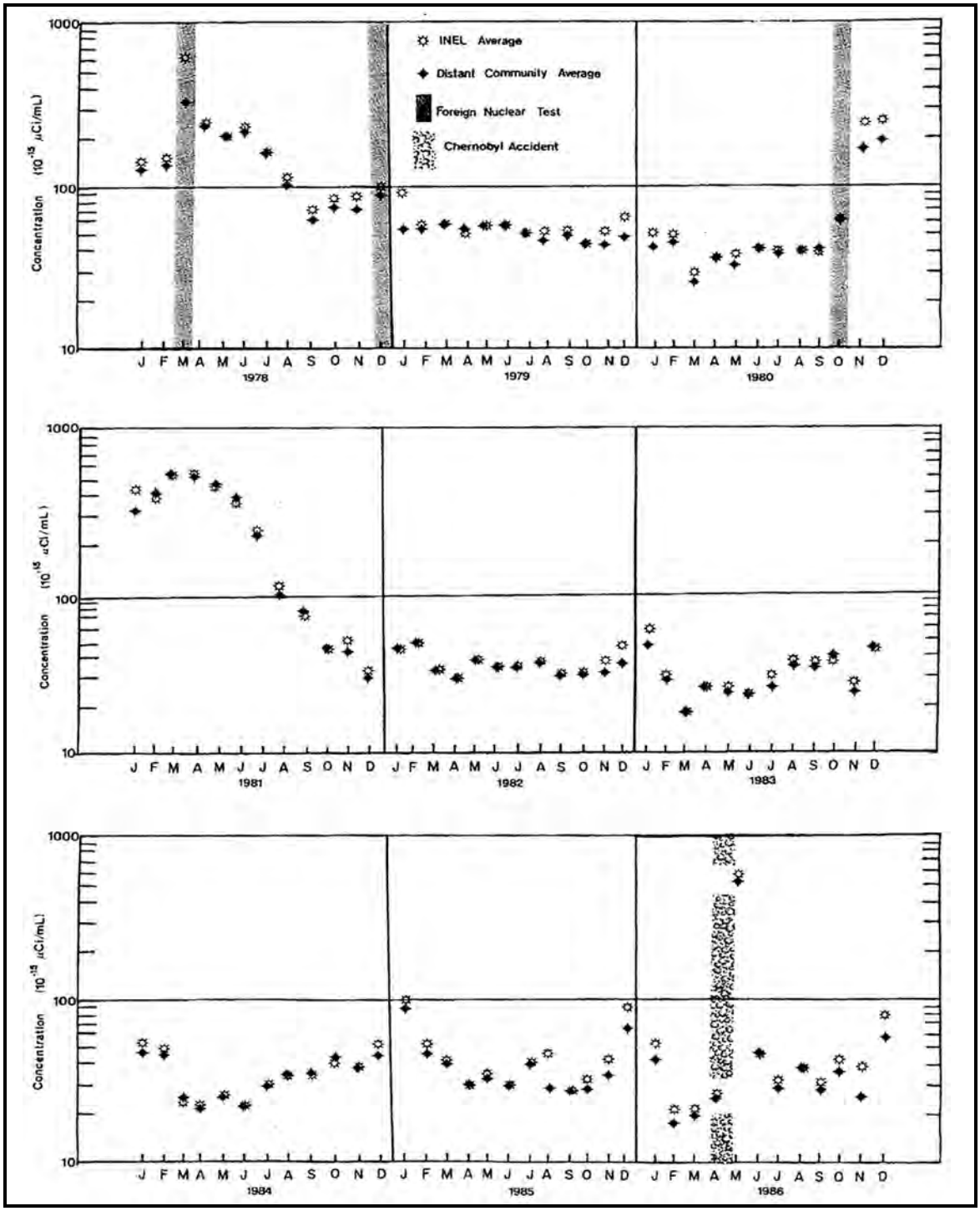

Figure 4-4. Onsite and distant particulate beta concentrations in air (Hoff, Chew, and Rope 1987, p. 17).

It is also interesting to note that the greater perturbations in the facility and distant community concentrations nearly all correlate with fallout from nuclear tests. Figure 4-5 (Williamson 1977a, p. 13) shows an example of such a perturbation, attributed to a September 26, 1976, atmospheric test conducted by the People's Republic of China, where the normal concentration was increased by a factor of 20 and had a 3-mo influence on the average concentration for all the air concentrations measured, on and off the site. In the history of air monitoring at the INL, an operational release has never approached the magnitude of perturbation created by fallout as illustrated in Figure 4-5.

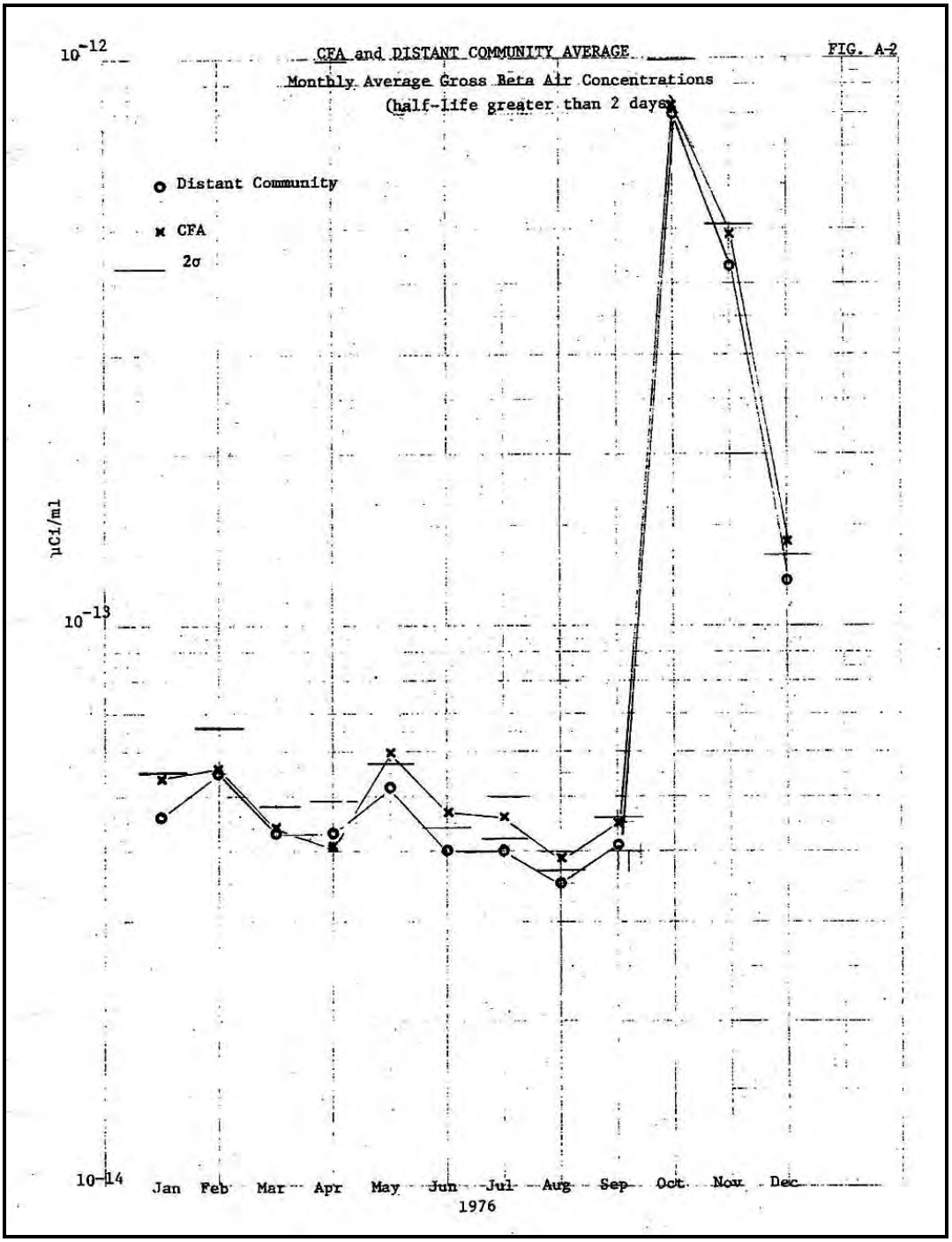

Figure 4-5. CFA air-monitoring (gross beta) data for 1976 (Williamson 1977a, p. 13).

4.2.2 Episodic Releases at INL

4.2.2.1 Stationary Low-Power Reactor No. 1 Accident

One significant accident at INL in the last 51 yr released substantial amounts of radioactive material to the environment. On January 3, 1961, a steam explosion at the SL-1 facility (near the present location of ARA II in Figure 4-1) killed three personnel and ruptured the SL-1 reactor vessel. The rupture 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 were carefully monitored and well-documented (Gammill 1961; Horan and Gammill 1961; Kunze 1962). All radiological doses to personnel involved in the rescue and the cleanup of the reactor building were carefully controlled and documented. The SL-1 accident did not affect any other INL facility with the effluent of radioactive material. The effluent traveled to the south of the facility, as shown in Figure 4-6 (DOE 1991).

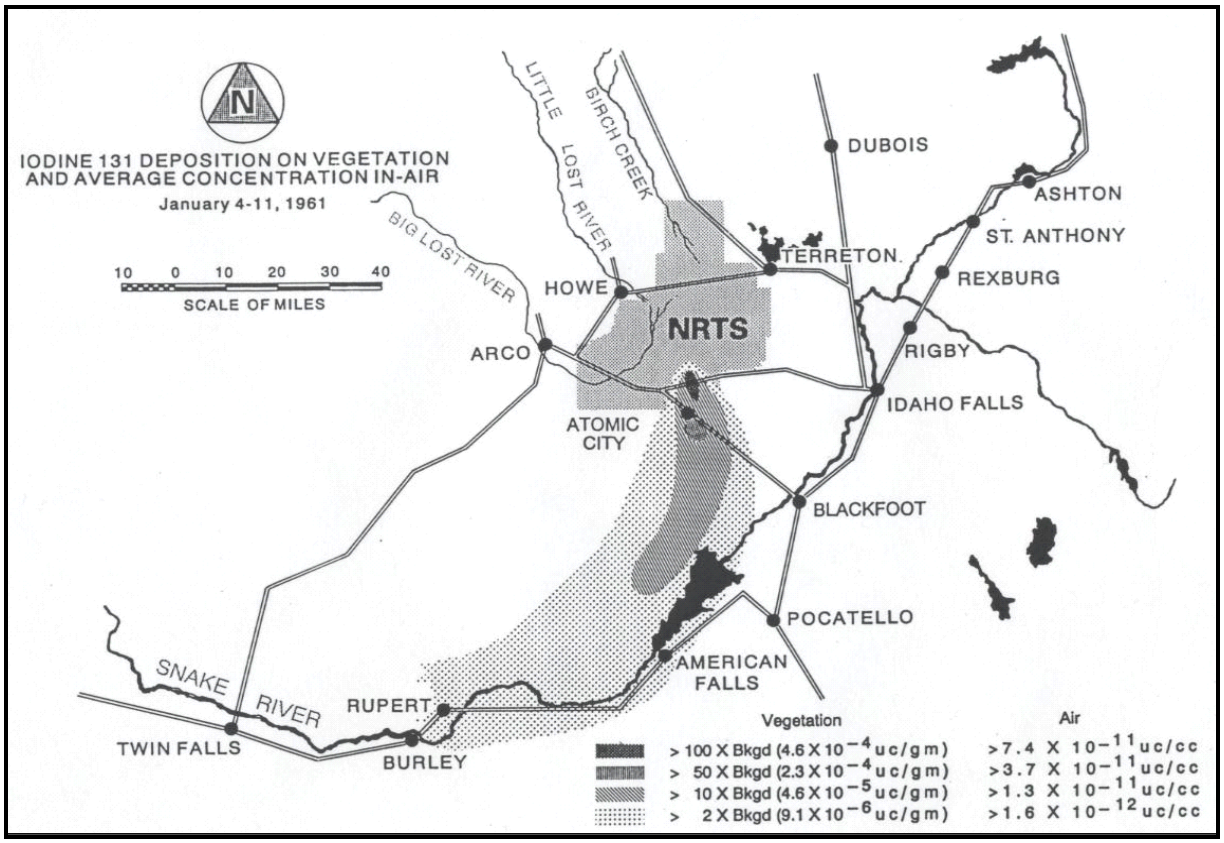

Figure 4-6. Dispersion coefficient contours for the SL-1 accident (redrafted from Horan and Gammill 1961).

4.2.2.2 Criticalities

Three accidental criticalities have occurred at the INTEC (formerly the ICPP). 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 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

INTEC and at CFA. A conservative analysis, described below, defined the potential radiological exposures that could have occurred to individuals at these locations.

4.2.2.2.1 INTEC Criticality of October 16, 1959

On October 16, 1959, at approximately 3:00 a.m., a criticality event occurred 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 χ/Q at 22 km matched the value calculated for Frenchman's Cabin (south of the INL, as shown on Figure 4-1) where offsite doses were calculated and reported in DOE (1991). RSAC-6 was used to calculate χ/Q values for the south end of the INTEC and CFA (Peterson 2004). These concentrations and intake quantities would be applicable only if the individual was in the respective areas on the morning of October 16, 1959. Table 4-10 lists the intakes applicable at the southern end of INTEC and CFA.

Table 4-10. Intakes (Bq/event) for criticalities that occurred at the INEEL [9].

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%

4.2.2.2.2 INTEC Criticality of January 25, 1961

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

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

As for the 1959 criticality, X/Q values were calculated for the southern end of INTEC and 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 INTEC and CFA if the individual was in the respective areas on January 25, 1961.

4.2.2.2.3 INTEC 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 1979a; Casto 1980). The fissioning lasted for about 20 min 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-s release of noble fission gases and a small fraction of the halogens produced in the event [10]. Table 4-10 lists the intakes applicable at the southern end of INTEC 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 Program. 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 General Electric Company literature (Linn 1962) about the test, the critical experiments and low-power testing of the assembly created insignificant radiological releases in comparison to the release from the 770-MW-s energy excursion that ended this series of tests. More information about this incident is included in Peterson (2004).

The released material for this test is consistent with the release modeled in DOE (1991). Table 4-10 lists the resultant inhaled quantities that would be applicable at the LPTF/Shield Test Pool Facility (STPF) location. The intakes are applicable only if the individual was at the 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 INL facilities [11]. Section 4.2.2.2 describes four of these events – the three criticalities at INTEC and the critical excursion in HTRE No. 3. The 12 remaining events are:

1. FEBT A
2. FEBT B
3. Fuel Element Cutting Facility (FECF)
4. 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 FEBT A at the Grid III location dictated that the offsite location of Reno Ranch, adjacent to the northern INL 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 can be drawn for

FEBT B, 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 FECF filter break at INTEC clearly contaminated an area south of INTEC. 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 INTEC and Frenchman's Cabin, a radiological impact analysis was conducted for 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 could also have affected INTEC, TRA, SPERT, the south INL Security Gate, or CFA. Because TRA and INTEC 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 and provided 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 hr and decayed for 70 d, began at 2:19 p.m. on March 20, 1957, at Grid III. The fuel element was assembled among other airframe parts to simulate an aircraft crash and burned for 2 hr using jet fuel as the burning agent. After 2 hr, the fuel element, which was intact, was believed not to have released much radioactive material (Brodsky and Beard 1960, p 42). Of the two FEBTs, Test A probably affected TAN more directly than Test B. Table 4-11 lists best estimates of the radionuclides and the intakes of those nuclides at TAN. 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 d (Brodsky and Beard 1960, p. 37).

After 90 s, most of the fuel element had melted and dispersed. DOE (1991, Figure 3-12) 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 $\times 10^{-14}$ hr/m^3 provided a concentration for the GE-ANP area in the range of 6×10^{-12} hr/m^3, the value used for determining the dose to 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 INTEC 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 Building CPP-601 for processing occurred in the FECF in Building 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, which resulted in the release of particulate activity to the south of INTEC.

Table 4-11. Intakes (Bq/event) for special tests at INL [12].

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
10/29–10/30/58	FECF Filter Break	South INTEC	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
		CFA	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

Approximately 100 Ci of long half-life particulate radioactive material was released over an area of approximately 200 acres (Horan 1959). The released radioactive material and quantities were the same as those published in DOE (1991). Table 4-11 lists best-estimate intakes of the radionuclides at the southern end of INTEC and at the CFA. 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 (Whitlow et al. 1959). A total of 100.25 hr was accumulated on the insert fuel cartridge at a maximum insert fuel temperature of approximately 2,500°F. The objectives of the test were (1) to 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 hr and (2) to measure the fission product release rate from uncoated fueled tubes as a function of temperature and operating time (Whitlow et al. 1959).

Table 4-12 lists the fission products released during the IET 14 test and the intakes at TRA/INTEC 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 the entire exposure period of 26 d.

Table 4-12. Intakes (Bq/event) for initial engine tests at INL [13].

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/INTEC	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/INTEC	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	South 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/INTEC	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	South 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/INTEC	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/INTEC	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)	South 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/INTEC	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

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). Data were obtained from this operation to evaluate:

- Endurance capabilities of the advanced metals at a design temperature of 2,000°F for extended periods (planned endurance testing to total 120 hr or more)
- The structural and metallurgical integrity of the fuel sheet in this particular cartridge design
- The nature and extent of fuel sheet damage, if any, and the effect on cartridge performance
- The performance potentials of the cartridge

The operation was successfully conducted to accumulate 80.75 hr at an insert extrapolated fuel sheet temperature of 2,015°F. The operation was terminated after 80.75 hr due to a release of fission products of such a quantity 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, but blisters were observed on the inner fuel sheets. In some instances the blisters had ruptured. The fission product release for this test was divided into two periods based on a review of effluent monitoring data. 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 (Evans 1959).

According to the meteorology of the testing period, the second period affected Frenchman's Cabin. Therefore, this analysis addressed the radiological impact on the TRA/INTEC and CFA. Table 4-12 lists the releases of fission products, which correspond with the Part B operation releases documented in DOE (1991) for the intakes applicable at TRA/INTEC 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 radiological doses in Table 4-12 are for the entire 9-d 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-hr of power (Highberg et al. 1959).

The fission product release for this test, which was modeled the same as that in DOE (1991), 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-hr, (2) a conservative particulate release of 14 Ci (1.5 Ci/hr for 9.5 hr), 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/hr 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 [14].

The release for this test series was modeled the same as that for DOE (1991) with Cerro Grande, south of the INL 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 INL South Security Gate just north of the junction of Highways 26 and 20 on the road to CFA, as shown in 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 when the reactor operated at power levels above 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 locations between November 2 and December 12, 1959. The radiological doses in Table 4-12 are for the entire exposure period of 40 d.

4.2.2.3.8 Initial Engine Test 18

IET 18, conducted between December 23, 1959, and February 8, 1960, and designated as the Phase II testing of the HTRE No. 3 engine, was an extension of the test program outlined for IET 16. The following is a 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).*
>
> *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* (Highberg et al. 1960).

In relation 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 INL 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 INL 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 operate the L2E-3 fuel cartridge at peak temperatures of 2,500°F and 2,600°F for 100 hr or more at each temperature level to evaluate the effectiveness of the zirconium dioxide coating against hydrolysis and the release of fission products
- To 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
- To 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-min-decayed curies. The total fission product release reported for IET 19 was 2,892 Ci. The release for this test was modeled like that for DOE (1991). Table 4-12 lists intakes for TRA/INTEC 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 radiological doses in Table 4-12 are for the entire exposure period of 13 d.

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 in IET 18. The test was conducted in the HTRE No. 3 reactor configuration. Releases of airborne radioactivity that corresponded 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 like that in DOE (1991) as a straight-line trajectory such that the centerline plume affected TRA/INTEC and CFA. The intakes for this test are listed in Table 4-12. An individual would have received intakes only if present at the locations between

November 22 and December 15, 1960. The tabulated intakes are for the entire exposure period of 24 d.

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 in IET 18 and the second part of the IET 25 test (Linn 1962; Highberg et al. 1961). The releases for the test correspond to 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 IET 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 INL South Security Gate. Table 4-12 lists the intakes that correspond 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 tabulated radiological intakes are for the entire exposure period of 16 d.

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 (Foster et al. 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 above 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 unfueled ceramic beryllium-oxide hexagonal tubes coated on the inner surface with zirconium dioxide (Foster et al. 1961).

The airborne release model was consistent with the model of DOE (1991) with an assumed straight-line trajectory between TAN and TRA/INTEC and CFA. The intakes are listed in Table 4-12 for TRA/INTEC and CFA. An individual would have been exposed to these intakes only if present at the locations between December 23 and December 28, 1960. The tabulated intakes are for the entire exposure period of 6 d.

4.3 EXTERNAL DOSE

External radiation dose at a facility can be created by direct radiation from two sources: (1) direct beta/gamma radiation from the facility or (2) gaseous effluents released from the facility or from adjacent facilities. In general, direct beta/gamma radiation from the facility increases with time because the general contamination of the area increases. In addition, as a facility ages, radioactive sources tend to accumulate at the facility, which causes the general background to increase with time. A responsible H&S organization observes and curbs such trends to prevent personnel exposures from increasing unnecessarily [15]. An excellent example of another aspect that can cause facility background to increase is the operational tests 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-in. duct from the HTRE engine to the stack was a significant radiation source. That is why operational facilities for that area were heavily shielded and personnel were required to be inside a shielded facility during a test. The following sections discuss facility fence-line film badge and TLD data that 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 Facility Fence-Line Annual Doses

Before 1970, many film badge or TLD measurements were made inside the INL boundary. During the IET period at TAN (1956 to 1961), many film badges were placed along the highways that triangulated the IET area and along highways at the southern end of the site. The badges were initially retrieved and read once a month. The frequency increased to 6 weeks in 1962 and changed back to monthly in 1963. Film badges were used through the first 9 mo of 1966, and TLDs were used from that time forward (AEC 1967a, p 8). Beginning in 1967, TLDs were changed on a semiannual basis. 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 (Dodd 1963) and that for the TLD was quoted as being 10 mrem when it was first used. With the annual background radiation field at the INL before the start of operations measured at 100 to 150 mrem/yr, the monthly value of 8 to 13 mrem is at the detection limit of the film badge or the TLD [16]. Therefore, the uncertainty for monthly change outs is higher than that for less frequent change outs.

Between the latter part of 1970 and the latter part of 1972, facility fence-line monitoring, facility locations, and facility arrangements had been established to provide a consistent location for TLD monitors (Walker 1973a, p 1-13). Although there have been minor facility changes since 1972, an effort was made to choose TLD locations that have been consistent from 1972 to the present. In most cases, a facility change did not result in location changes for TLDs. The TRA facility did have minor changes over time but the majority of the TLD locations were consistent. TRA initially had an “L” shape, but in 1992, the facility plot changed to a square.

There are two areas at TRA where TLDs were used especially to track gamma fields for work planning: (1) the North Storage Area (NSA), and (2) the TRA Warm Waste Ponds (WWPs). The NSA was designated for storage of mildly activated or contaminated reactor hardware. It was just outside and north of the TRA perimeter, but still within the area fence and away from normally inhabited buildings or areas. TLDs 7, 8, and 9 were placed around the NSA to track area radiation fields due to the irradiated material. The TRA WWPs were outside the TRA to the east. The elevation of the ponds was at least 5 ft below the elevation of the TRA facility. TLDs 3, 4, and 5 were placed around the WWPs to track radiation fields from the ponds for personnel control and exposure reasons. In the original attempt to characterize the TRA radiation field, TLDs 3, 4, 5, 7, 8, and 9 were excluded. The final characterization used TLDs 1, 7, 12, and 13, which increased the gamma field for TRA by a factor of 3 to 4 above the initial value. Row 2 of Table 4-13 lists area TLD numbers.

From 1972 through 1983, facility fence-line TLD measurements, made on a 6-mo basis with five TLDs at each facility position, are available in the quarterly Environmental Monitoring Data reports (Walker 1972, 1973a,b, 1974a,b, 1975a,b,c; Dahl 1975; Sill 1976; Bills 1976; Williamson 1976a,b, 1977a,b,c, 1978a,b, DOE 1978b,c,d, 1979b,c,d,e, 1980 b,c,d,e, 1981b,c,d,e, 1982b,c,d,e, 1983b,c,d,e, 1984). Figure 4-7 (from Williamson 1976a) shows that uncertainty can vary from less than 10% to 20% for a given set of readings. At each of the 34 monitoring locations, there are normally five TLDs for a potential 170 readings for a 6-mo period. For this particular 2-yr set of data, 1.5% of the 136 readings were assigned a 2-sigma uncertainty of 16 to 20% and 18.4% of the readings were assigned a 2-sigma uncertainty of 11% to 15%. However, 80% of the 136 values ascribed for the 34 locations over the 2-yr period have a 2-sigma uncertainty of less than 10% [17].

To supply facility values for the period from 1952 to 1972, the highest 6-mo value from April 1972 to April 1973 for a facility was multiplied by 2 and applied to each year between 1952 and 1972 [18] (Not

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 onsite TLD monitoring data (Williamson 1976a, p. 38).

all the listed facilities commenced operation or even existed in 1952.). Facility fence-line TLD measurements could not be found for 1984 through 1992, but for 1993 and beyond such measurements were included in the EMRs (Mitchell 1994; Mitchell, Peterson, and Hoff 1995; Mitchell et al. 1996, 1997; Evans et al. 1998; Saffel et al. 2000; Stoller 2002a,b,c, 2003). For the period from 1984 to 1992 when TLD measurements were missing, reasonable interpolations were used to provide the missing values. In addition, background TLD measurements that correspond with the facility fence-line 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 (AEC 1971,

1972, 1973,1974; ERDA 1975, 1976, 1977b; DOE 1978a, 1979a, 1980a, 1981a, 1982a, 1983a, Hoff, Chew, and Dickson 1984; Mitchell 1994; Mitchell, Peterson, and Hoff 1995; Mitchell et al. 1996, 1997; Evans et al. 1998; Saffel et al. 2000, Stoller 2002a,b,c, 2003). A more detailed discussion of the data is included in Peterson (2004).

Table 4-13. INL facility fence direct gamma values (TLD – background) (mR) [19].

	ARA I & II	SPERT	TAN-TSF	TAN-LOFT	TAN-LPTF	CFA	TRA	INTEC	RWMC	EBR-II	TREAT	Back-ground
	TLDs used											
Year	1, 2, 3	All	All	All	All	All	1, 7, 12, 13	2, 11, 12, 13	13, 15, 17, 19	All	All	
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
2003	43	21	0	8	0	11	31	15	14	20	11	117
2004	24	11	6	7	1	7	26	13	19	23	10	118
2005	22	13	3	8	4	12	23	20	10	26	24	120

4.3.2 Facility Air Immersion Doses

INL facility air immersion (beta-gamma) doses could be calculated from the noble gas and halogen portions of the operational releases and, if applicable, from the noble gas portion of the applicable episodic releases. This calculation should be unnecessary because these releases would be recorded in the fence-line TLD doses listed in Table 4-13.

4.4 UNCERTAINTY

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

Operational Releases. Discussions with the authors of DOE (1991) suggest that operational releases, which were monitored, could be low by a factor of not more than 2 [20]. When annual normalized ground-level concentration values are applied to operational releases, the uncertainty could be increased. However, in considering this increased uncertainty, it is interesting to note the facility air-monitoring results in the annual EMRs. In each case, the facility air concentration is compared to that for a distant community, usually Idaho Falls. The concentration is normally indistinguishable from the concentration for the distant community, as discussed in Section 4.2.1 [21].

Episodic Releases. As described in DOE (1991), episodic releases are maximum reasonable values based on the amount of material available to be released and the conditions of the test. For such releases, the inhaled quantities (in becquerels) were maximized by assuming the downwind exposed individual was subjected to the plume centerline concentration for the total time, night and day in most cases, of the release. In spite of the effort to be reasonably conservative in exposure estimates, some of the authors of DOE (1991)have stated that the release considered for a particular episodic event might be low by as much as a factor of 3 [22].

A concerted effort has been made to reduce the number of radionuclides involved in releases for episodic events. Overall, the mix of radionuclides for all the episodic events is complicated by the type (aged versus fresh) and the relative quantities of each. When viewed together, the episodic events can be categorized into three categories: (1) criticalities that involve fresh fission products that have relatively short half-lives in comparison to radionuclides released from the FEBTs, for example; (2) releases involving long half-life, aged fission products (FEBTs and the release from the FECF Filter Break); and (3) releases from the remaining IET tests that released short half-life radionuclides, which are generally characterized as fresh fission products, and long half-life radionuclides, which are characterized as aged fission products. 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 calculated for that location [23].

During 1988, a minor contamination incident occurred at the RWMC that led to minor inhalation exposures to a few individuals, but that incident has not been incorporated into the environmental doses. The following paragraph from the 1988 EMR (Hoff, Mitchell, and Moore 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, the uncertainty of individual measurements made with film badges and TLDs can be as high as ±100% depending on the frequency of change out (i.e., once per month, which was generally the case with film badges). The data for 1952 to 1972 in Table 4-13 are based on the highest 6-mo TLD values from 1972 for the facility. Although the GE-ANP IETs were conducted in the late 1950s and early 1960s (IET 26, the last, ended on March 31, 1961), tests with planned releases were administratively and meteorologically controlled so the airborne effluent traveled to the northeast over the monitoring grid and so that adjacent facilities were not affected. In spite of these controls, the 1952 to 1962 values for the TAN areas (TSF, LOFT, and LPTF) could be low by a factor of 3 [24]. However, after 1967 when facility fence-line measurements were routinely made with five TLDs at a given location, the uncertainty is generally ascribed at less than 10% or about 13 mrem. Less than about 20% of the time, these measurements have an ascribed level of uncertainty as high as 20% [25].

Dose reconstruction for an individual whose location is unknown should use the intakes listed in Table 4-3 (CFA) and the exposures for INTEC listed in Table 4-13. The suggested values maximize the resultant individual dose.

4.5 ATTRIBUTIONS AND ANNOTATIONS

Where appropriate in this document, bracketed callouts have been inserted to indicate information, conclusions, and recommendations to assist in the process of worker dose reconstruction. These callouts are listed in this section with information that identifies the source and justification for each item. Conventional references are provided in the next section that link data, quotations, and other information to documents available for review on the Oak Ridge Associated Universities (ORAU) Team servers.

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

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

[1] Peterson, Henry. ORAU Team. Site Expert. September–December 2003.
Although major events at one facility have sometimes been measured at other nearby facilities, they have never necessitated emergency measures.

[2] Peterson, Henry. ORAU Team. Site Expert. September-December 2003.
Tests were conducted over the GRID III when the meteorological conditions guaranteed the effluent would not affect other INL facilities. The monitoring grid was directed to the northeast, a direction that was not toward any other facility at the INL Site.

[3] Peterson, Henry. ORAU Team. Site Expert. September-December 2003.
Figure 4-3 shows a concentration isopleth of about 5 for Idaho Falls compared to over 100 for all site facilities.

[4] Peterson, Henry. ORAU Team. Site Expert. September-December 2003.
This is a recommendation to use the absorption type that is predominant based on information on chemical form and solubility type found in Peterson 2004.

[5] Peterson, Henry. ORAU Team. Site Expert. September-December 2003.
This is a recommendation to use the absorption type that will give the largest dose.

[6] Peterson, Henry. ORAU Team. Site Expert. September-December 2003.
An example of such analysis is Ebersole (1956).

[7] Peterson, Henry. ORAU Team. Site Expert. September-December 2003.
This limited list of radionuclides provides more than 95% of the effective dose equivalent based on ICRP Publication 68 (ICRP 1994). The complete list of radionuclides from Peterson (2004) was multiplied by the effective dose coefficient for inhalation of 5-µm activity median aerodynamic diameter (AMAD) from ICRP Publication 68 and then added together. In steps, nuclides were eliminated based on effective dose being less than 0.1% of the total, then 0.3%, and then 0.4%. The fraction of effective dose equivalent lost is less than 5%, but the activity lost is considerably more.

[8] Peterson, Henry. ORAU Team. Site Expert. September-December 2003.
The default geometric standard deviation of 3 has been confirmed by phone conversation and e-mail from C. Bloom to N. Rohrig of November 9, 2006.

[9] Peterson, Henry. ORAU Team. Site Expert. September-December 2003.
This limited list of radionuclides provides more than 95% of the effective dose equivalent based on ICRP Publication 68 (ICRP 1994). The complete list of radionuclides from Peterson (2004) was multiplied by the effective dose coefficient for inhalation of 5-µm AMAD from ICRP Publication 68 and then added together. In steps, nuclides were eliminated based on effective dose being less than 0.1% of the total, then 0.3%, and then 0.4%. The fraction of effective dose equivalent lost is less than 5%, but the activity lost is considerably more.

[10] Peterson, Henry. ORAU Team. Site Expert. September-December 2003.
These values are generally consistent with reactor safety licensing assumptions.

[11] Peterson, Henry. ORAU Team. Site Expert. September-December 2003.
When the analysis of DOE (1991) was completed, the meteorological diffusion trajectories were reviewed to determine which INL facilities were affected. The result of these trajectory reviews was that only 16 releases had the potential to affect other INL facilities.

[12] Peterson, Henry. ORAU Team. Site Expert. September-December 2003.
This limited list of radionuclides provides more than 95% of the effective dose equivalent based on ICRP Publication 68 (ICRP 1994). The complete list of radionuclides from Peterson (2004) was multiplied by the effective dose coefficient for inhalation of 5-µm AMAD from ICRP Publication 68 and then added together. In steps, nuclides were eliminated based on effective dose being less than 0.1% of the total, then 0.3%, and then 0.4%. The fraction of effective dose equivalent lost is less than 5%, but the activity lost is considerably more.

[13] Peterson, Henry. ORAU Team. Site Expert. September-December 2003.
This limited list of radionuclides provides more than 95% of the effective dose equivalent based on ICRP Publication 68 (ICRP 1994). The complete list of radionuclides from Peterson (2004) was multiplied by the effective dose coefficient for inhalation of 5-µm AMAD from ICRP Publication 68 and then added together. In steps, nuclides were eliminated based on effective dose being less than 0.1% of the total, then 0.3%, and then 0.4%. The fraction of effective dose equivalent lost is less than 5%, but the activity lost is considerably more.

[14] Peterson, Henry. ORAU Team. Site Expert. September-December 2003.
These values are generally consistent with reactor safety licensing assumptions.

[15] Peterson, Henry. ORAU Team. Site Expert. September-December 2003.
As a facility ages, dirt and waste, including radioactive material, generally accumulate in the facility. A good example of such accumulation is at a foundry or machine shop. Knowledgeable management will remove these materials and wastes as they accumulate; if not, the general background radiation, as well as clutter of the facility will increase with time.

[16] Peterson, Henry. ORAU Team. Site Expert. September-December 2003.
The uncertainty of 10 mrem is roughly the same as the expected result of 8 to 13 mrem.

[17] Peterson, Henry. ORAU Team. Site Expert. September-December 2003.
This comes from counting the values in Figure 4-7.

[18] Peterson, Henry. ORAU Team. Site Expert. September-December 2003.
This explains how missing data were generated.

[19] Peterson, Henry. ORAU Team. Site Expert. September-December 2003.
See the references listed in the last paragraph of Section 4.3.1.

[20] Peterson, Henry. ORAU Team. Site Expert. September-December 2003.
Discussion between Henry Peterson, Doug Wenzel, and Richard Dickson in June 2003.

[21] Peterson, Henry. ORAU Team. Site Expert. September-December 2003.
These results are shown in the environmental monitoring reports, for example Walker (1972).

[22] Peterson, Henry. ORAU Team. Site Expert. September-December 2003.
Discussion between Henry Peterson, Doug Wenzel, and Richard Dickson in June 2003.

[23] Peterson, Henry. ORAU Team. Site Expert. September-December 2003.
This limited list of radionuclides provides more than 95% of the effective dose equivalent based on ICRP Publication 66 (ICRP 1994). The complete list of radionuclides from Peterson (2004) was multiplied by the effective dose coefficient for inhalation of 5-µm AMAD from Appendix B of ICRP Publication 66 and then added together. In steps, nuclides were eliminated based on effective dose being less than 0.1% of the total, then 0.3%, and then 0.4%. The fraction of effective dose equivalent lost is less than 5%, but the activity lost is considerably more.

[24] Peterson, Henry. ORAU Team. Site Expert. September-December 2003.
This follows from the discussion of episodic releases in this uncertainty section. Such releases primarily affected these areas.

[25] Peterson, Henry. ORAU Team. Site Expert. September-December 2003. This is based on Figure 4-7, which is a typical set of results.

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