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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.
01/08/2010	02	Revised to combine ANL-W TBD into the INL TBD. Updated Intake and External Dose tables to add data from 2006 to 2008. Added ^{129}I intake values to intake tables for all areas. Incorporates formal internal review comments. No changes were needed as a result of formal NIOSH review. Constitutes a total rewrite of the document. Training required: As determined by the Objective Manager. Initiated by Jo Ann 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
CERT	Controlled Environmental Radioiodine (later Release) Test
CFA	Central Facilities Area
CFR	Code of Federal Regulations
Ci	curie
CPP	Chemical Processing Plant
d	day
DOE	U.S. Department of Energy
EBR	Experimental Breeder Reactor
ECF	Expended Core Facility
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)
GCRE	Gas-Cooled Reactor Experiment
GSD	geometric standard deviation
H&S	Health and Safety (division)
hr	hour
HTRE	Heat Transfer Reactor Experiment
ICPP	Idaho Chemical Processing Plant
ICRP	International Commission on Radiological Protection
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, including ANL-W
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
mL	milliliter
mo	month
mR	milliroentgen
mrem	millirem
MW	megawatt
NRF	Naval Reactors Facility
NCRP	National Council on Radiation Protection and Measurements
NIOSH	National Institute for Occupational Safety and Health
NOAA	National Oceanic and Atmospheric Administration
NRTS	National Reactor Testing Station
NSA	North Storage Area
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
SRDB Ref ID	Site Research Database Reference Identification
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
WERF	Waste Experimental Production Facility
WWP	Warm Waste Pond
yr	year
ZPPR	Zero Power Plutonium (later Physics) Reactor
α	alpha activity
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, 42 C.F.R. Pt. 82) define "performance of duty" for DOE employees with a covered cancer or restrict the "duty" to nuclear weapons work (NIOSH 2007).

The statute also 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 excludes Naval Nuclear Propulsion Facilities from being covered under the Act, 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 occupational radiation exposures are considered valid for inclusion in a 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 (NIOSH 2007):

- Background radiation, including radiation from naturally occurring radon present in conventional structures

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

- Radiation from X-rays received in the diagnosis of injuries or illnesses or for therapeutic reasons

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)] and at the Argonne National Laboratory – West (ANL-W). 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 INL facility and the ANL-W. The releases discussed here have been divided into (1) normal "chronic" operational releases and (2) episodic releases that generally have been of short duration (DOE 1991). 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 *fenceline* 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 fenceline doses (minus background) are presented for 11 locations. This TBD presents ANL-W facility fenceline doses (minus background) for the Environmental Breeder Reactor No. I (EBR-I) location for 1952 to 1972 and the Transient Reactor Test (TREAT) facility and EBR-II locations, for 1972 to the present.

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

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).

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

southeast corner of INL. Since the mid-1970s, essentially all ANL-W operations have been conducted at the present location, as shown in Figure 4-1.

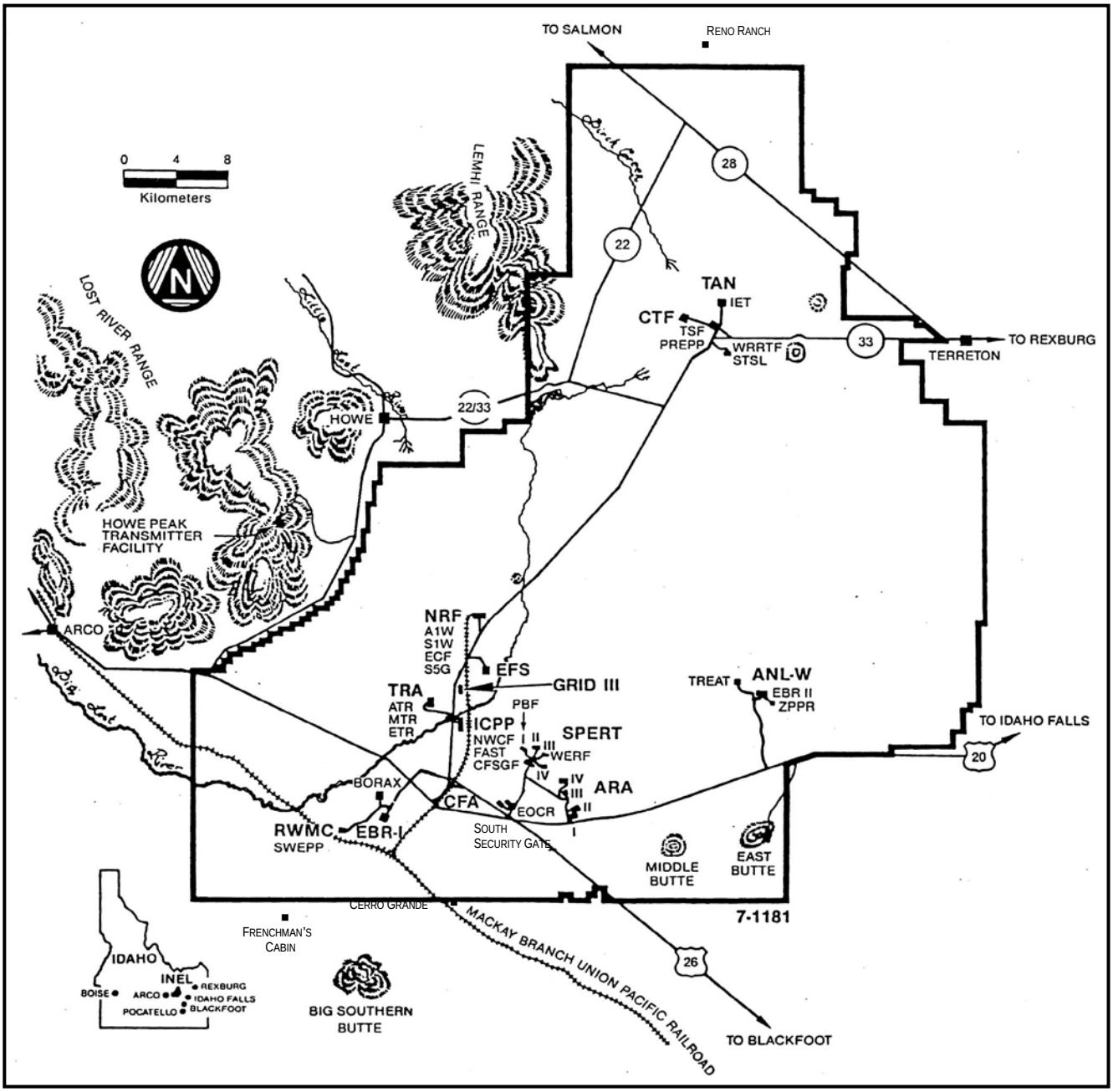

Figure 4-1. INL site map (Hoff, Chew, and Rope 1987, p. 46).

During the 50-year 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) EBR-II, at ANL-W at the southeast 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 (Knecht et al. 1997, p 4). Throughout its history, the Chem Plant, or CPP as it is commonly known, has reprocessed fuel from test reactors at INL, zirconium-clad fuel reclaimed from various reactors, stainless-steel-clad fuel from EBR-II, and many AEC test reactors from around the world. Apart from the GE-ANP Program, which is 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. Calculations performed for this TBD show that although ICPP airborne effluent is the most radiologically significant release at INL, the impact to all facility workers is significantly below the allowable and acceptable limits (Peterson 2004). Figure 4-1 shows the locations of INL facilities.

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-year history of the site has demonstrated that the large expanse of INL and this meteorological characteristic have been satisfactorily effective in maintaining the operational isolation of each facility [1].

Beginning with the GE-ANP Program, which began in the early 1950s, the 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 1988, 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 (RSAC-4) program (Wenzel 1990). The results of the study were published in *Idaho National Engineering Laboratory Historical Dose Evaluation* (DOE 1991). All releases considered for that report are the bases 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 program (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 program, see Peterson (2004).

Figure 4-2 shows the chronology of facilities and programs at 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 the map into context with the timeline in Figure 4-2.

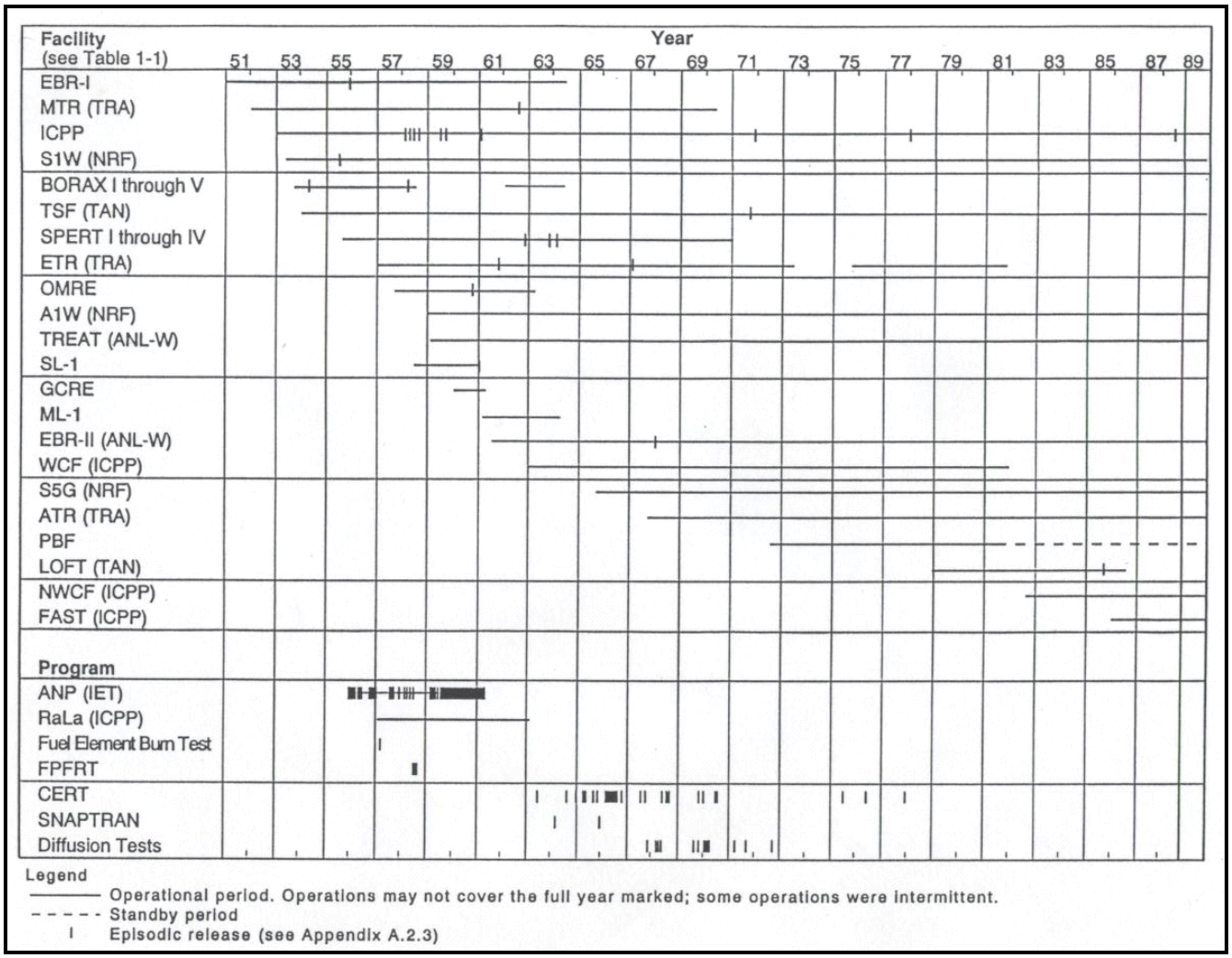

Figure 4-2. Chronology of facilities and programs at the INL site (DOE 1991).

EBR-I was the first reactor to operate at the site. It and the Boiling-Water Reactor Experiments (BORAX-I through -V), in the southwest corner of the site, 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 EBR-II, the TREAT Facility, ZPPR, 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 (GCRE), 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 (NRF), 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 INL 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-year average of mesoscale dispersion isopleths of ground-level air concentrations (DOE 1991) shown in Figure 4-3 was used.[2] For 1993 to 2008, 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, 2008) were used to calculate the facility annual concentration.

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

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

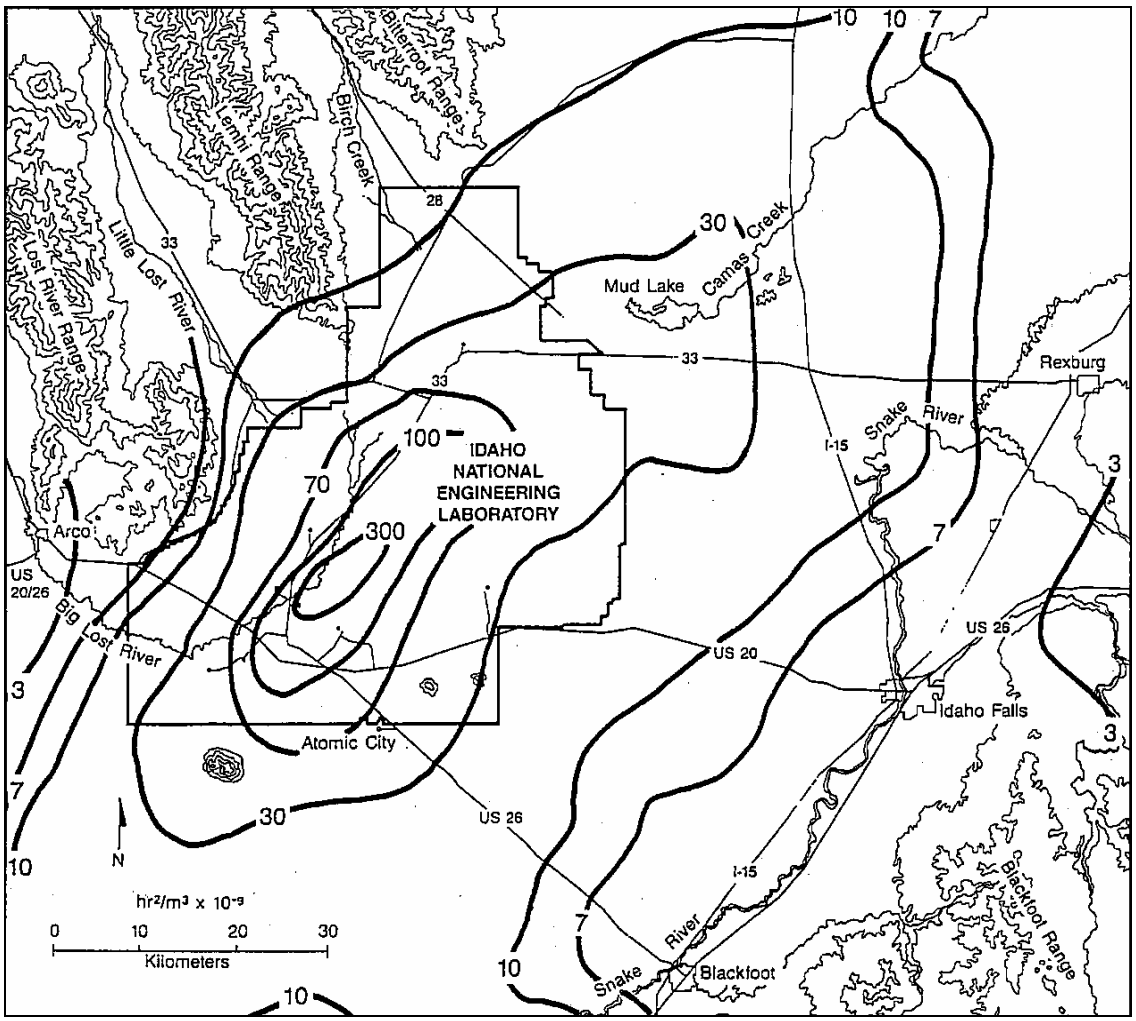

Figure 4-3. Nine-year (1974 to 1983 excluding 1978) average mesoscale dispersion isopleths of air concentration at ground level ($hr^2/m^3 \times 10^{-9}$), normalized to unit release rate (DOE 1991).

Reactor Research Test Facility, Process Experimental Pilot Plant, and Low Power Test Facility (LPTF), depending on the year of operation (ORAUT 2007). The SPERT area includes the Waste Experimental Reduction Facility (WERF) 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, 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.

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 2006 through 2008 values are the 95th

percentile of the average of the most recent 10 years of data. In order to address thyroid cancers, I-129 environmental intakes have been added. Iodine-129 values were calculated by multiplying the 95th percentile ratio of ^{129}I to ^{90}Sr stack release data (1,800) by the ^{90}Sr intake. 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.

Table 4-1. Intake (Bq/yr) by year for ANL-W, 1952 to 2008 [6].

Nuclide	Ce-144	I-131	Pm-147	Pu-238	Pu-239, -240	Ru-106	Sr-89	Sr-90	Y-91	I-129
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	2.3E+02
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	1.2E+03
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	2.2E+03
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	3.1E+03
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	3.2E+03
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	5.2E+03
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	7.4E+03
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	6.1E+03
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	3.8E+02
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	4.5E+02
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	6.8E+02
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	5.0E+03
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	1.3E+03
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	4.9E+03
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	1.4E+03
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	3.8E+02
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	2.2E+03
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	6.5E+02
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	4.9E+02
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	2.0E+03
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	5.0E+02
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	1.1E+02
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	6.7E+01
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	3.4E+01
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	7.0E-01
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	9.0E+00
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	3.4E+00
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	1.6E+01
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	7.7E-01
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	5.9E-01
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	5.0E-01
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	2.0E-01
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	2.3E-01
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	1.2E+00
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	2.9E-02
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	4.1E-02
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	4.9E-02
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	1.3E-02

Nuclide	Ce-144	I-131	Pm-147	Pu-238	Pu-239, -240	Ru-106	Sr-89	Sr-90	Y-91	I-129
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	3.6E-04
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	1.7E-01
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	1.5E-02
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	6.8E-02
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	1.4E-01
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	4.1E-03
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	2.0E-03
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	4.3E-02
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	2.0E-02
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	7.9E-03
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	6.3E+00
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	2.2E-01
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	6.8E+00
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	0.0E+00
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	2.0E+00
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	2.3E+01
2006–2008	3.19E-6	4.36E-3	4.27E-3	1.08E-5	4.57E-5	8.55E-9	6.76E-5	2.32E-3	0.00E+0	4.2E+00

Table 4-2. Intake (Bq/yr) by year for ARA, 1952 to 2008 [6].

Nuclide	Ce-144	I-131	Pm-147	Pu-238	Pu-239, 240	Ru-106	Sr-89	Sr-90	Y-91	I-129
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	3.24E+02
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	1.75E+03
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	3.06E+03
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	4.32E+03
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	4.50E+03
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	7.38E+03
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	1.06E+04
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	8.82E+03
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	5.40E+02
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	6.48E+02
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	9.72E+02
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	7.20E+03
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	1.80E+03
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	7.02E+03
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	1.98E+03
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	5.40E+02
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	3.06E+03
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	9.18E+02
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	6.84E+02

Nuclide	Ce-144	I-131	Pm-147	Pu-238	Pu-239, 240	Ru-106	Sr-89	Sr-90	Y-91	I-129
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	2.88E+03
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	7.20E+02
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	3.78E+01
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	1.98E+02
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	5.04E+01
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	9.90E-01
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	1.30E+01
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	5.04E+00
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	6.84E+01
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	3.24E+00
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	2.52E+00
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	5.04E-01
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	2.70E-01
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	3.24E-01
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	1.19E+00
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	1.24E-01
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	4.14E-02
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	4.86E-02
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	5.76E-02
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	1.19E-03
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	5.76E-01
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	5.04E-02
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	6.84E-01
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	6.12E-01
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	9.72E-03
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	4.50E-03
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	1.03E-01
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	4.50E-02
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	2.70E-02
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	2.16E+01
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	7.02E-01
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	1.60E+01
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	0.00E+00
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	7.20E-01
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	2.34E+01
2006–2008	1.22E-6	4.11E-3	2.49E-3	2.14E-5	4.68E-5	8.55E-9	6.77E-5	3.14E-3	0.00E+0	5.65E+00

Table 4-3. Intake (Bq/yr) by year for CFA, 1952 to 2008 [6].

Nuclide	Ce-144	I-131	Pm-147	Pu-238	Pu-239, 240	Ru-106	Sr-89	Sr-90	Y-91	I-129
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	9.7E+02
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	5.2E+03
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	9.2E+03
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	1.3E+04
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	1.4E+04
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	2.2E+04
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	3.2E+04
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	2.7E+04
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	1.6E+03
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	2.0E+03
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	2.9E+03
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	2.2E+04
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	5.4E+03
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	2.2E+04
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	5.9E+03
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	1.6E+03
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	9.4E+03
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	2.7E+03
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	2.0E+03
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	8.6E+03
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	5.0E+02
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	1.1E+02
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	6.7E+01
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	3.4E+01
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	7.0E-01
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	9.0E+00
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	3.4E+00
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	1.6E+01
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	7.7E-01
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	5.9E-01
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	5.0E-01
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	2.0E-01
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	2.3E-01
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	1.2E+00
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	2.9E-02
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	4.1E-02
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	4.9E-02
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	1.3E-02

Nuclide	Ce-144	I-131	Pm-147	Pu-238	Pu-239, 240	Ru-106	Sr-89	Sr-90	Y-91	I-129
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	3.6E-04
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	1.7E-01
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	1.5E-02
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	6.8E-02
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	1.4E-01
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	4.1E-03
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	2.0E-03
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	4.3E-02
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	2.0E-02
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	7.9E-03
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	6.3E+00
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	2.2E-01
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	2.3E+01
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	0.0E+00
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	2.9E+00
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	5.2E+01
2006–2008	4.96E-6	1.15E-2	1.69E-2	1.90E-5	9.82E-5	1.78E-8	1.44E-4	5.36E-3	0.00E+0	9.6E+00

Table 4-4. Intake (Bq/yr) by year for INTEC, 1952 to 2008 [6].

Nuclide	Ce-144	I-131	Pm-147	Pu-238	Pu-239, 240	Ru-106	Sr-89	Sr-90	Y-91	I-129
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	9.7E+02
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	5.2E+03
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	9.2E+03
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	1.3E+04
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	1.4E+04
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	2.2E+04
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	3.2E+04
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	2.7E+04
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	1.6E+03
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	2.0E+03
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	2.9E+03
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	2.2E+04
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	5.4E+03
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	2.2E+04
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	5.9E+03
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	1.6E+03
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	9.4E+03
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	2.7E+03
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	2.0E+03

Nuclide	Ce-144	I-131	Pm-147	Pu-238	Pu-239, 240	Ru-106	Sr-89	Sr-90	Y-91	I-129
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	8.6E+03
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	2.2E+03
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	2.7E+02
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	6.7E+02
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	1.5E+02
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	7.0E+00
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	3.8E+01
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	1.5E+01
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	6.8E+01
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	3.2E+00
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	2.5E+00
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	7.2E-01
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	8.1E-01
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	9.9E-01
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	2.7E+00
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	1.2E-01
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	1.7E-01
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	2.2E-01
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	5.8E-02
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	1.2E-03
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	5.8E-01
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	1.5E-01
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	6.8E-01
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	6.1E-01
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	1.4E-02
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	6.5E-03
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	1.5E-01
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	2.0E-01
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	7.9E-02
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	2.2E+01
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	7.0E-01
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	6.8E+01
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	0.0E+00
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	3.6E+00
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	1.0E+02
2006–2008	6.02E-6	2.03E-2	2.49E-2	3.74E-5	1.93E-4	3.74E-8	2.85E-4	1.16E-2	0.00E+0	2.1E+01

Table 4-5. Intake (Bq/yr) by year for RWMC, 1952 to 2008 [6].

Nuclide	Ce-144	I-131	Pm-147	Pu-238	Pu-239, 240	Ru-106	Sr-89	Sr-90	Y-91	I-129
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	2.3E+02
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	1.2E+03
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	2.2E+03
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	3.1E+03
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	3.2E+03
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	5.2E+03
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	7.4E+03
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	6.1E+03
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	3.8E+02
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	4.5E+02
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	6.8E+02
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	5.0E+03
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	1.3E+03
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	4.9E+03
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	1.4E+03
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	3.8E+02
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	2.2E+03
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	6.5E+02
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	4.9E+02
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	2.0E+03
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	5.0E+02
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	2.7E+02
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	2.0E+02
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	3.4E+01
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	9.9E-01
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	9.0E+00
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	3.4E+00
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	1.6E+01
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	7.7E-01
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	5.9E-01
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	5.0E-01
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	2.0E-01
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	3.2E-01
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	4.0E-01
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	4.1E-02
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	5.8E-02
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	2.2E-01
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	2.0E-02

Nuclide	Ce-144	I-131	Pm-147	Pu-238	Pu-239, 240	Ru-106	Sr-89	Sr-90	Y-91	I-129
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	1.2E-03
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	5.8E-01
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	5.0E-02
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	6.8E-01
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	6.1E-01
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	1.4E-02
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	4.5E-03
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	1.0E-01
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	4.5E-02
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	1.8E-02
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	2.2E+01
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	7.0E-01
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	2.3E+01
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	0.0E+00
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	2.7E+00
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	2.3E+01
2006–2008	4.96E-6	1.15E-2	1.69E-2	1.90E-5	9.82E-5	1.78E-8	1.44E-4	5.36E-3	0.00E+0	9.6E+00

Table 4-6. Intake (Bq/yr) by year for SPERT, 1952 to 2008 [6].

Nuclide	Ce-144	I-131	Pm-147	Pu-238	Pu-239, 240	Ru-106	Sr-89	Sr-90	Y-91	I-129
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	9.7E+02
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	5.2E+03
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	9.2E+03
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	1.3E+04
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	1.4E+04
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	2.2E+04
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	3.2E+04
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	2.7E+04
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	1.6E+03
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	2.0E+03
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	2.9E+03
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	2.2E+04
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	5.4E+03
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	2.2E+04
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	5.9E+03
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	1.6E+03
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	9.4E+03
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	2.7E+03
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	2.0E+03

Nuclide	Ce-144	I-131	Pm-147	Pu-238	Pu-239, 240	Ru-106	Sr-89	Sr-90	Y-91	I-129
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	8.6E+03
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	2.2E+03
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	3.8E+01
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	4.7E+02
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	1.5E+02
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	9.9E+00
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	3.8E+01
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	1.5E+01
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	2.3E+02
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	3.2E+00
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	2.5E+00
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	7.2E-01
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	8.1E-01
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	9.9E-01
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	1.2E+00
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	1.2E-01
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	1.7E-01
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	7.0E-02
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	5.8E-02
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	1.2E-03
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	5.8E-01
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	1.5E-01
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	6.8E-01
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	6.1E-01
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	1.4E-02
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	4.5E-03
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	1.0E-01
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	4.5E-02
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	7.9E-02
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	2.2E+01
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	7.0E-01
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	1.6E+01
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	0.0E+00
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	1.2E+00
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	1.7E+01
2006–2008	1.95E-6	3.52E-3	4.98E-3	2.09E-5	3.55E-5	6.24E-9	4.80E-5	2.80E-3	0.00E+0	5.0E+00

Table 4-7. Intake (Bq/yr) by year for TAN, 1952 to 2008 [6].

Nuclide	Ce-144	I-131	Pm-147	Pu-238	Pu-239, 240	Ru-106	Sr-89	Sr-90	Y-91	I-129
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	2.3E+02
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	1.2E+03
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	2.2E+03
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	3.1E+03
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	3.2E+03
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	5.2E+03
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	7.4E+03
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	6.1E+03
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	3.8E+02
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	4.5E+02
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	6.8E+02
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	5.0E+03
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	1.3E+03
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	4.9E+03
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	1.4E+03
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	3.8E+02
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	2.2E+03
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	6.5E+02
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	4.9E+02
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	2.0E+03
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	5.0E+02
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	1.1E+02
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	6.7E+01
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	1.5E+01
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	9.9E-01
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	1.3E+00
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	3.4E+00
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	6.8E+00
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	3.2E-01
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	2.5E-01
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	2.2E-01
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	8.1E-02
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	9.9E-02
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	1.2E-01
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	1.2E-02
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	4.1E-02
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	2.2E-02
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	1.3E-02

Nuclide	Ce-144	I-131	Pm-147	Pu-238	Pu-239, 240	Ru-106	Sr-89	Sr-90	Y-91	I-129
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	3.6E-04
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	1.7E-01
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	1.5E-02
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	6.8E-02
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	1.4E-01
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	9.7E-03
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	2.0E-03
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	4.3E-02
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	2.0E-02
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	7.9E-03
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	6.3E+01
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	2.2E+00
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	6.8E+00
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	0.0E+00
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	3.6E-01
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	5.2E+00
2006–2008	6.03E-7	3.59E-3	1.69E-3	6.36E-5	5.11E-5	1.78E-9	1.44E-5	6.14E-3	0.00E+0	1.1E+01

Table 4-8. Intake (Bq/yr) by year for TRA, 1952 to 2008 [6].

Nuclide	Ce-144	I-131	Pm-147	Pu-238	Pu-239, 240	Ru-106	Sr-89	Sr-90	Y-91	I-129
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	9.7E+02
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	5.2E+03
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	9.2E+03
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	1.3E+04
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	1.4E+04
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	2.2E+04
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	3.2E+04
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	2.7E+04
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	1.6E+03
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	2.0E+03
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	2.9E+03
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	2.2E+04
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	5.4E+03
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	2.2E+04
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	5.9E+03
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	1.6E+03
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	9.4E+03
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	2.7E+03
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	2.0E+03

Nuclide	Ce-144	I-131	Pm-147	Pu-238	Pu-239, 240	Ru-106	Sr-89	Sr-90	Y-91	I-129
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	8.6E+03
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	2.2E+03
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	2.7E+02
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	2.0E+02
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	3.4E+01
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	3.1E+00
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	1.3E+01
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	1.5E+01
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	1.6E+01
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	7.7E-01
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	5.9E-01
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	5.0E-01
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	2.0E-01
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	2.3E-01
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	4.0E-01
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	4.1E-02
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	1.7E-01
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	2.2E-01
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	5.8E-02
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	1.2E-03
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	5.8E-01
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	5.0E-02
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	6.8E-01
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	2.0E-01
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	9.7E-03
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	4.5E-03
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	1.0E-01
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	4.5E-02
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	1.8E-02
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	2.2E+01
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	7.0E-01
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	6.8E+01
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	0.0E+00
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	3.6E+00
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	5.2E+01
2006–2008	6.03E-6	1.01E-2	1.26E-2	2.70E-5	9.98E-5	1.78E-8	1.44E-4	7.99E-3	0.00E+0	1.4E+01

Environmental air sampling at the facility areas has been performed at least since the mid-1950s where airborne effluents were known or suspected to exist [7]. The early IDO Health and Safety (H&S) Division EMRs (AEC 1960, 1961a,b, 1962a,b, 1963, 1964a,b, 1965a,b, 1966a,b, 1967a,b, 1968a,b, 1969a,b, 1970a,b) 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 reported alpha, beta, and ^{131}I 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, 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 (GSD) of 5.3. Nevertheless, this comparison provides confidence in the results of Table 4-1 through 4-8. The value at the upper 84% confidence value, 1.9 (0.36 × 5.3), is less than the default GSD of 3 assumed for environmental results [8].

Figure 4-4 shows the variation of the INL environmental monitoring sampling results for the 9-year 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-year period has not differed from the distant community concentrations by more than a factor of 2. Inspection of subsequent 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).

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-month influence on the average concentration for all the air concentrations measured, on and off the site. In the history of air monitoring at INL, an operational release has never approached the magnitude of perturbation created by fallout.

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 years 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 the cloud traveled from the reactor building were carefully monitored and well-documented (Gammill 1961; Horan and Gammill 1961; Kunze 1962). All

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, p. 14	0.18	0.01[a]
	β	0.64 pCi/m^3	AEC 1969a, p. 14	56	337[a]
	I-131	<0.08 pCi/m^3	AEC 1969a, p. 14	<7.1	1.4[a]
1969	α	0.003 pCi/m^3	AEC 1970a, p. 14	0.27	2.4E-3[a]
	β	0.501 pCi/m^3	AEC 1970a, p. 14	44	118[a]
	I-131	<0.014 pCi/m^3	AEC 1970a, p. 14	<1.2	2.1[a]
1970	Gross β	0.6 pCi/m^3	AEC 1971, p. 5	53	74[a]
	Max. gr. β at CFA	0.81 pCi/m^3	AEC 1971, p. 5	72	74[a]
1973	Gross β	95 ± 42 fCi/m^3	AEC 1974, p. 14	8.4 ± 3.7	0.8[a]
EBR-I	Sr-90	3.4 ±3.0 fCi/m^3	AEC 1974, p. 16	0.3 ± 0.27	0.15[b]
	Nb-95	1.0-2.5 fCi/m^3	AEC 1974, p. 16	0.09–0.22	---
	Cs-137	7-17 fCi/m^3	AEC 1974, p. 16	0.6–1.5	---
	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, Chew, and Rope 1987	3,290 ± 1330	890[e]
1988	Kr-85 at CFA	108 ± 69 pCi/m^3	Hoff, Chew, and Moore 1989	9,770 ± 6130	14,000[e]
1990	Kr-85 at CFA	27.7 pCi/m^3	Hoff et al. 1992	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.

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).

4.2.2.2 Criticalities

Three accidental criticalities have occurred at INTEC. The first occurred on October 16, 1959; the second on January 25, 1961; and the third on October 17, 1978. The 1978 criticality released essentially just the noble fission gases produced during the criticality; halogens and solids were removed by the 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 south end of INTEC and at CFA and at the EBR-I facility, because that facility was postulated to be in the plume path. A conservative analysis, described below, defined the potential radiological exposures that could have occurred to individuals at these locations. An additional critical excursion occurred in 1958 as part of the critical experiments and low-power testing phase of the HTRE No. 3 reactor configuration for the GE-ANP Program.

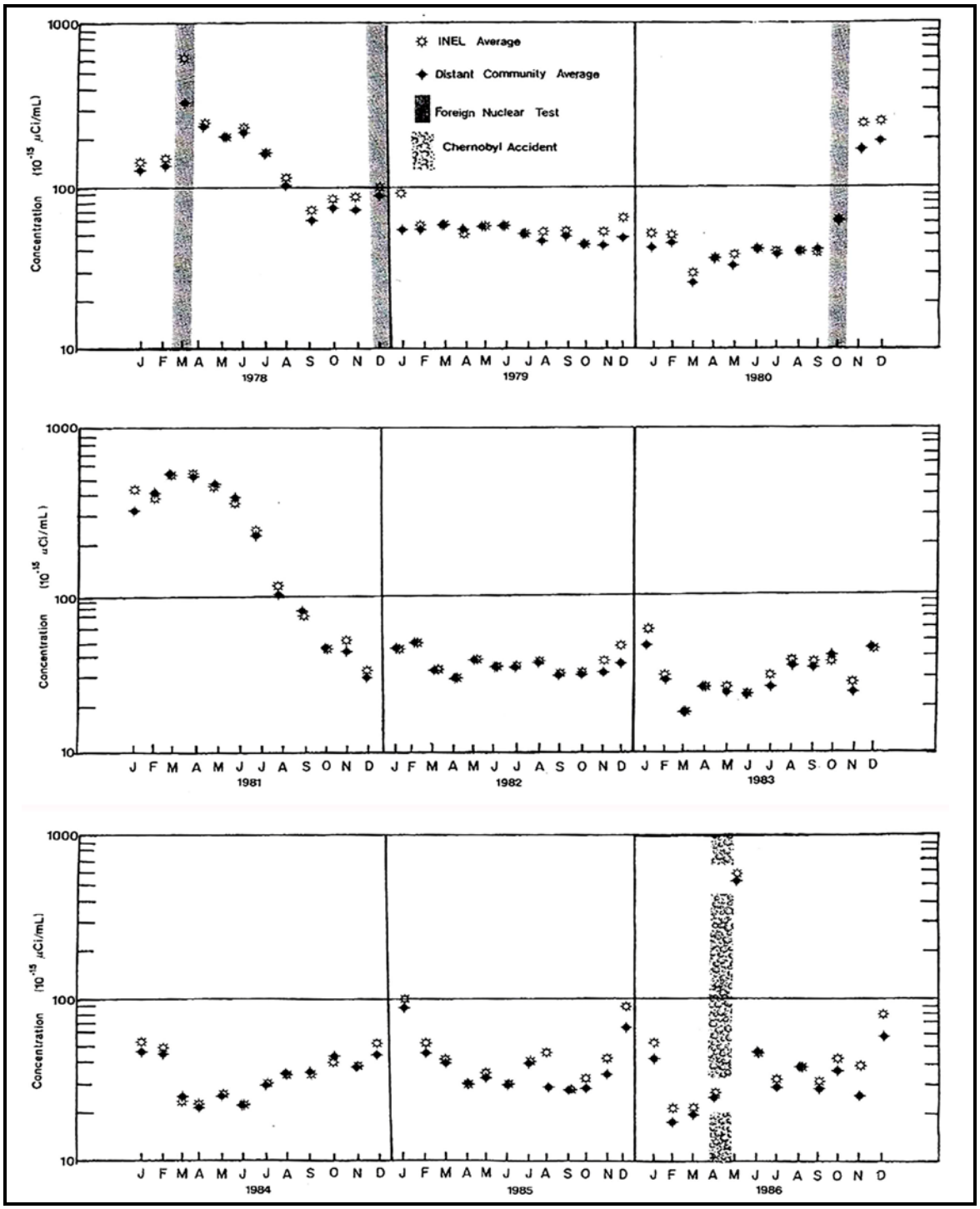

Figure 4-4. Onsite and distant particulate beta concentrations in air (Hoff, Chew, and Rope 1987, p. 17).

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

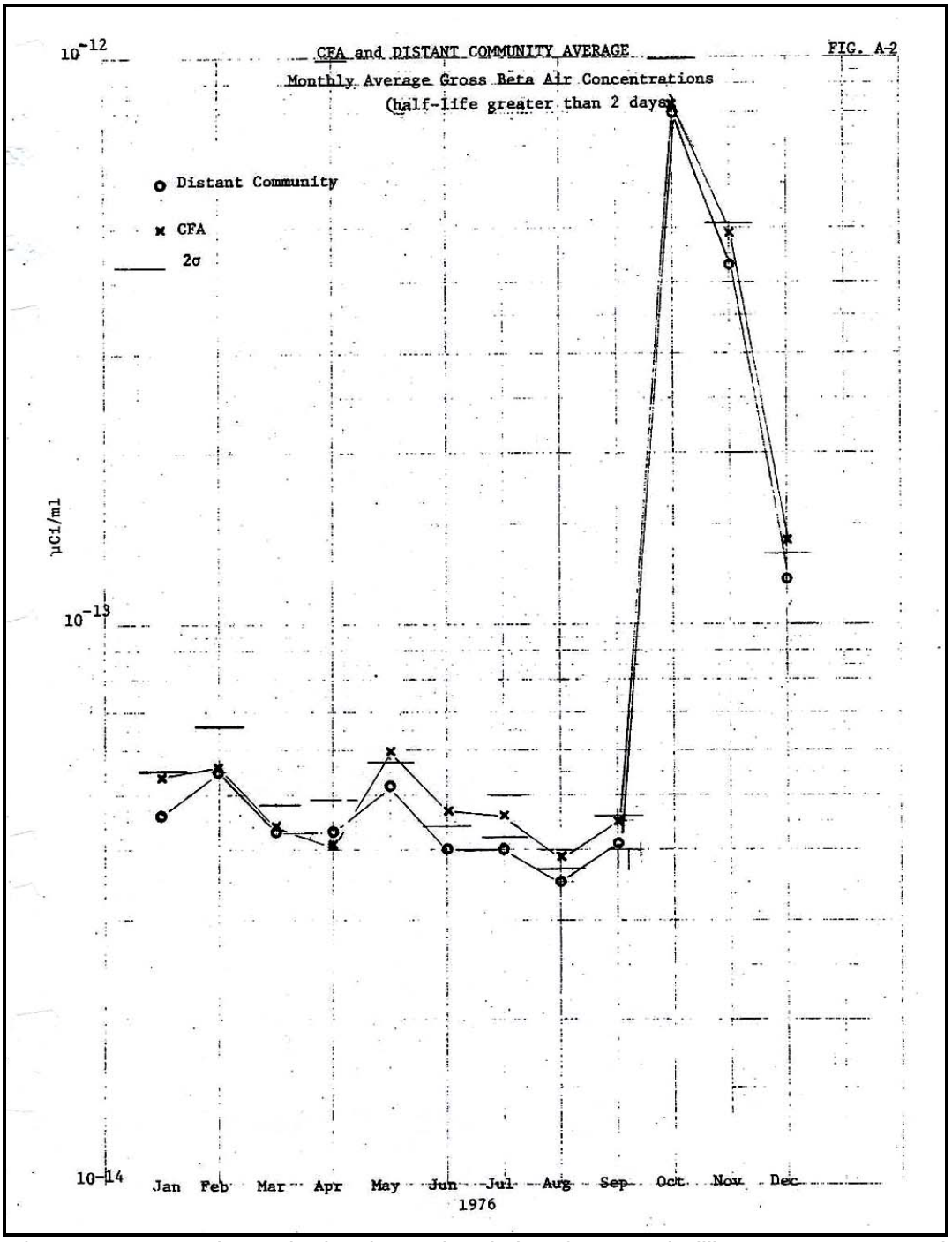

Figure 4-5. CFA air-monitoring (gross beta) data for 1976 (Williamson 1977a, p. 13).

conditions were modeled so that the χ/Q at 22 km matched the value calculated for Frenchman's Cabin (south of INL, as shown on Figure 4-1), where offsite doses were calculated and reported in DOE (1991). The RSAC-6 computer program was used to calculate χ/Q values for the south end of 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 south end of INTEC, CFA, and EBR-1.

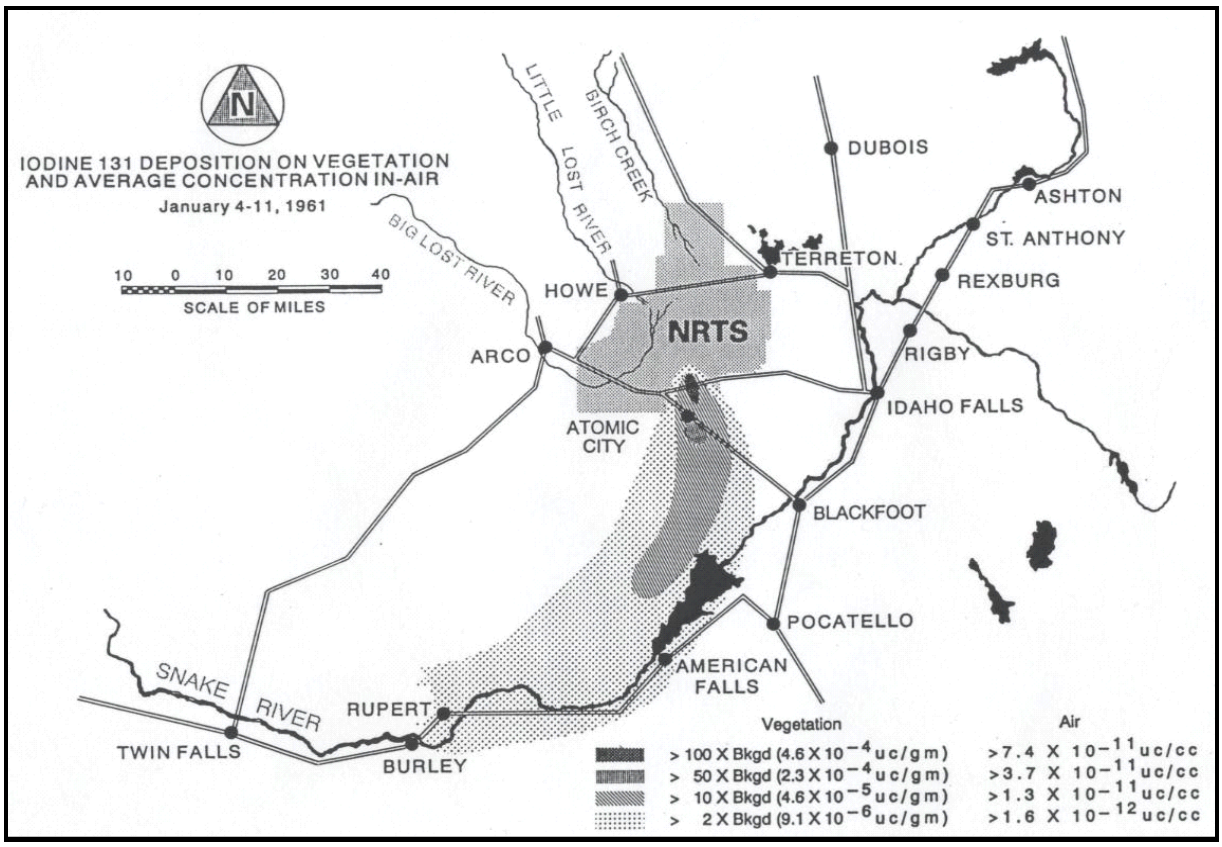

Figure 4-6. Dispersion coefficient contours for the SL-1 accident (redrafted from Horan and Gammill 1961).

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 stated:

> *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 south end of INTEC, CFA, and EBR-1. 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 south end of INTEC, CFA, and EBR-1 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 minutes and involved about 3.0×10^{18} fissions. This event did not result in significant radiation exposures to personnel, and there was no contamination of general plant areas.

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

Date	11/18/1958	10/16/1959	10/16/1959	10/16/1959	1/25/1961	1/25/1961	1/25/1961	10/17/1978	10/17/1978
Event	IET 13	Criticality	Criticality	Criticality	Criticality	Criticality	Criticality	Criticality	Criticality
Exposure location	LPTF/STPF	South ICPP	CFA area	EBR-I area	South ICPP	CFA area	EBR-I area	South ICPP	CFA area
Rb-89	8.1E+2	5.2E+5	8.8E+4	2.1E+4	1.5E+1	1.0E+0	4.4E-1	2.2E+3	3.7E+4
Sr-91	4.1E+2	5.4E+4	9.2E+3	2.2E+3	1.7E+2	1.1E+1	4.8E+0	1.7E+1	7.0E+2
Sr-92	1.9E+2	6.5E+4	1.1E+4	2.6E+3	1.5E+2	9.7E+0	4.3E+0	5.6E+0	2.1E+2
Y-92	5.6E+1	7.5E+3	1.3E+3	3.1E+2	7.3E+1	4.9E+0	2.2E+0	9.1E-2	2.6E+1
Y-93	2.0E+1	5.9E+3	1.0E+3	2.4E+2	1.9E+1	1.3E+0	5.7E-1	1.3E-1	1.5E+1
Te-133								1.1E+3	8.0E+3
I-131(elem.)	3.4E+0	2.9E+2	5.0E+1	1.2E+1	1.9E+0	1.3E-1	5.7E-2	3.9E-2	1.7E+0
I-133	7.1E+1	6.2E+3	1.1E+3	2.6E+2	3.8E+1	2.5E+0	1.1E+0	4.4E+0	5.8E+2
I-134	2.0E+2	4.6E+4	7.8E+3	1.9E+3	7.2E+1	4.8E+0	2.1E+0	1.1E+2	3.4E+3
I-135	4.6E+2	2.0E+4	3.4E+3	8.2E+2	1.1E+2	7.2E+0	3.2E+0	5.5E+1	2.2E+3
Cs-138	5.0E+3	7.7E+5	1.3E+5	3.1E+4				5.2E+2	5.6E+4
Ba-139	3.4E+3	4.0E+5	6.8E+4	1.6E+4	6.6E+2	4.4E+1	1.9E+1	1.4E+2	1.8E+4
La-141	2.0E+2	2.8E+4	4.7E+3	1.1E+3	1.0E+2	6.8E+0	3.0E+0	4.6E-1	9.0E+1
La-142	9.3E+1	2.3E+4	3.9E+3	9.4E+2	4.3E+1	2.8E+0	1.2E+0	4.8E-1	5.7E+1
U-234	7.1E-2								
Dose fract	94.4%	93.9%	92.5%		94.5%	94.5%		98.3%	97.1%

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 south 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 about the test (Linn 1962), 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, CFA, or EBR-1. 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 hours and decayed for 70 days, 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 hours using jet fuel as the burning agent. After 2 hours, 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 days (Brodsky and Beard 1960, p. 37).

After 90 seconds, most of the fuel element had melted and dispersed. DOE (1991, Figure 3-12) shows the trajectory of the plume, and Figure 3-13 in that document 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 FECF 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.

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 and EBR-1. 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 No. 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 hours

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
		EBR-I Area	6.3E-1	6.3E-1	1.4E+0	2.0E+0	1.3E-1	6.6E-1	4.2E-12	9.0E+0	2.6E-6

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 hours 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 IET 14 and the intakes at TRA/INTEC, CFA, and EBR-1. 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 intakes are for the entire exposure period of 26 days.

4.2.2.3.5 Initial Engine Test 15(B)

IET 15 was conducted at TAN between May 27 and June 24, 1959. This test involved the evaluation of the L2C-1 insert cartridge, which was of the concentric ring design. The fuel sheet was made of a chromium-uranium dioxide-titanium core clad with an iron-chromium-yttrium alloy (Evans 1959). 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 hours 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 hours at an insert extrapolated fuel sheet temperature of 2,015°F. The operation was terminated after 80.75 hours due to a release of fission products of such a quantity 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. The first period – 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, EBR-1, 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, EBR-1, and CFA. An individual would have been exposed to these concentrations and intakes only if present at the locations between June 16 and June 24, 1959. The radiological intakes in Table 4-12 are for the entire 9-day exposure period.

4.2.2.3.6 Initial Engine Test 16

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

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

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 hours), and (3) a conservative release fraction of 3.0×10^{-7} for iodine isotopes, the highest fraction measured during the test. To preserve these values and arrive at release fractions for other groups of radionuclides, engineering judgment and preliminary data from a few iterations of the RSAC program were used. To meet the stated criteria of 1.5 Ci/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(B) 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 intakes in Table 4-12 are for the entire exposure period of 40 days.

4.2.2.3.8 Initial Engine Test 18

IET 18, conducted between December 23, 1959, and February 8, 1960, 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(A), 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 hours 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-minute-decayed curies. The total fission product release reported for IET 19(A) was 2,892 Ci. The release for this test was modeled like that for DOE (1991). Table 4-12 lists intakes for TRA/INTEC, EBR-1, 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 intakes in Table 4-12 are for the entire exposure period of 13 days.

4.2.2.3.10 Initial Engine Test 25(A)

IET 25(A), performed between November 15 and December 19, 1960, was an extension of the Phase II testing program in IET 18. The test was conducted in the HTRE 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, EBR-1, 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 days.

4.2.2.3.11 Initial Engine Test 25(B)

IET 25, performed between November 15 and December 19, 1960, was an extension of the Phase II testing program in IET 18 and the second part of IET 25 (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 days.

4.2.2.3.12 Initial Engine Test 26(A)

IET 26(A), 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, TRA/INTEC, EBR-1, and CFA. The intakes are listed in Table 4-12 for TRA/INTEC, EBR-1, 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 days.

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 fenceline 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 Fenceline 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 south 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 months 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 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 fenceline 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. No film badge or TLD information is available for the EBR-I and BORAX locations when they were operational.

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 fenceline TLD measurements, made on a 6-month basis with five TLDs at each facility position, are available in the quarterly Environmental Monitoring Data reports (Walker 1972, 1973a,b, 1974a,b, 1975a,b,c; Dahl 1975; Sill 1976; Bills 1976; Williamson 1976a,b, 1977a,b,c, 1978a,b; DOE 1978b,c,d, 1979b,c,d,e, 1980b,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-month period. For this particular 2-year set of data, 1.5% of the 136 readings were assigned a 2-sigma uncertainty of 16 to 20% and 18.4% of the readings were

Table 4-13. INL facility fence direct gamma values (TLD–background) (mR) [17].

Year	ARA I & II	SPERT	TAN-TSF	TAN-LOFT	TAN-LPTF	CFA	TRA	INTEC	RWMC	EBR-II	TREAT	Background
	TLDs used											
	1, 2, 3	All	All	All	All	All	1, 7, 12, 13	2, 11, 12, 13[a]	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
2006	4	7	0	0	0	7	21	36	3	6	2	122
2007	2	8	0	0	0	1	23	40	6	6	2	130

a. All after 2005.

assigned a 2-sigma uncertainty of 11% to 15%. However, 80% of the 136 values ascribed for the 34 locations over the 2-year period have a 2-sigma uncertainty of less than 10% [18].

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).

To supply facility values for the period from 1952 to 1972, the highest 6-month value from April 1972 to April 1973 for a facility was multiplied by 2 and applied to each year between 1952 and 1972 [19]. (Not all the listed facilities had begun operation or even existed in 1952). Facility fenceline TLD measurements could not be found for 1984 through 1992, but for 1993 and beyond such measurements were included in the EMRs (Mitchell 1994; Mitchell, Peterson, and Hoff 1995; Mitchell et al. 1996, 1997; Evans et al. 1998; Saffel et al. 2000; Stoller 2002a,b,c, 2003). For the period from

1984 to 1992 when TLD measurements were missing, reasonable interpolations were used to provide the missing values. In addition, background TLD measurements that correspond with the facility fenceline TLD measuring periods were recorded in the EMRs. All reduced facility fenceline TLD data (facility fenceline data minus background) in the EMRs are listed in Table 4-13 (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, 2007, 2008). A more detailed discussion of the data is included in Peterson (2004).

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 fenceline 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 IETs that released short-half-life radionuclides, which are generally characterized as fresh fission products, and long-half-life radionuclides, which are characterized as aged fission products. 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].

In 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-month 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 fenceline 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 the area with the highest annual dose for each year of employment 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 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 on 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 health physics program policies, procedures, or practices related to atomic weapons activities at the

site. As mentioned above, 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 with an isopleth 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.
This limited list of radionuclides provides more than 95% of the effective dose equivalent based on International Commission on Radiological Protection (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.

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

[8] Peterson, Henry. ORAU Team. Site Expert. September-December 2003.
The default GSD of 3 has been confirmed by telephone conversation and e-mail from C. Bloom of the ORAU Team to N. Rohrig on 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.
See the references listed in the last paragraph of Section 4.3.1.

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

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

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

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

[22] Peterson, Henry. ORAU Team. Site Expert. September-December 2003.
Discussion between Henry Peterson, Doug Wenzel (INL), and Richard Dickson (INL) 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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GLOSSARY

absorption
In external dosimetry, process in which radiation energy is imparted to material. In internal dosimetry, movement of material to blood regardless of mechanism.

absorption type
Categories for materials according to their rate of absorption from the respiratory tract to the blood, which replaced the earlier inhalation clearance classes. Defined by the International Commission on Radiological Protection, the absorption types are F: deposited materials that are readily absorbed into blood from the respiratory tract (fast solubilization), M: deposited materials that have intermediate rates of absorption into blood from the respiratory tract (moderate rate of solubilization), and S: deposited materials that are relatively in the respiratory tract (slow solubilization). Also called solubility type.

activity
Amount of radioactivity. The International System unit of activity is the becquerel (1 disintegration per second); the traditional unit is the curie [37 billion (3.7×10^{10}) becquerels].

activity median aerodynamic diameter (AMAD)
Diameter of a unit density sphere with the same terminal settling velocity in air as that of the aerosol particle whose activity is the median for the entire aerosol.

alpha radiation
Positively charged particle emitted from the nuclei of some radioactive elements. An alpha particle consists of two neutrons and two protons (a helium nucleus) and has an electrostatic charge of +2.

becquerel (Bq)
International System unit of radioactivity equal to 1 disintegration per second; 1 curie equals 37 billion (3.7×10^{10}) Bq.

beta dose
Designation (i.e., beta) on some records for external dose from beta and less-energetic X-ray and gamma radiation, often for shallow dose or dose to the lens of the eye.

beta radiation
Charged particle emitted from some radioactive elements with a mass equal to 1/1,837 that of a proton. A negatively charged beta particle is identical to an electron. A positively charged beta particle is a positron.

breeder reactor
Nuclear reactor that makes more new fissionable material than it consumes.

contamination
Radioactive material in an undesired location including air, soil, buildings, animals, and persons.

decontamination
Reduction or removal of radioactive material from a structure, area, object, or person. Decontamination can occur through (1) treating the surface to remove or decrease the contamination or (2) allowing natural radioactive decay to occur over a period of time.

dose

In general, the specific amount of energy from ionizing radiation that is absorbed per unit of mass. Effective and equivalent doses are in units of rem or sievert; other types of dose are in units of roentgens, rads, reps, or grays.

dose equivalent

In units of rem or sievert, product of absorbed dose in tissue multiplied by a weighting factor and sometimes by other modifying factors to account for the potential for a biological effect from the absorbed dose. See *dose*.

dose reconstruction (DR)

Process of analyzing the available information including evaluation of historical methods and data to estimate the dose a person could have received from one or more radiation exposures.

dosimeter

Device that measures the quantity of received radiation, usually a holder with radiation-absorbing filters and radiation-sensitive inserts packaged to provide a record of absorbed dose received by an individual. See *film dosimeter* and *thermoluminescent dosimeter*.

dosimetry

Measurement and calculation of internal and external radiation doses.

effective dose equivalent (EDE)

Average of the dose equivalents weighted for the susceptibility of harm to different tissues or organs in units of rem or sievert. See *dose* and *weighting factor*.

effluent

Liquid or gaseous waste released to the environment.

Energy Employees Occupational Illness Compensation Program Act of 2000, as amended (EEOICPA; 42 U.S.C. § 7384 *et seq.*)

Law that provides for evaluation of cause and potential compensation for energy employees who have certain types of cancer.

exposure

(1) In general, the act of being exposed to ionizing radiation. (2) Measure of the ionization produced by X- and gamma-ray photons in air in units of roentgens.

external dose

Dose received from radiation emitted by sources outside the body.

external radiation

Radiation from sources outside the body.

film

(1) In the context of external dosimetry, radiation-sensitive photographic film in a light-tight wrapping. See *film dosimeter*. (2) X-ray film.

film badge

See *film dosimeter*.

film dosimeter

Package of film for measurement of ionizing radiation exposure for personnel monitoring purposes. A film dosimeter can contain two or three films of different sensitivities, and it can contain one or more filters that shield parts of the film from certain types of radiation. When developed, the film has an image caused by radiation measurable with an optical densitometer. Also called film badge.

fissile

Capable of undergoing nuclear fission by capturing thermal (slow) neutrons. The three primary fissile materials are ^{233}U, ^{235}U, and ^{239}Pu. Fissile generally indicates spontaneous fission.

fission

Splitting of the nucleus of an atom (usually of a heavy element) into at least two other nuclei and the release of a relatively large amount of energy. This transformation usually releases two or three neutrons.

fuel

Fissile material in a nuclear reactor that is the source of the chain reaction of neutrons that produce fission and release energy as heat and other electromagnetic radiation.

gamma radiation

Electromagnetic radiation (photons) of short wavelength and high energy (10 kiloelectron-volts to 9 megaelectron-volts) that originates in atomic nuclei and accompanies many nuclear reactions (e.g., fission, radioactive decay, and neutron capture). Gamma photons are identical to X-ray photons of high energy; the difference is that X-rays do not originate in the nucleus.

gamma ray, particle, or photon (γ)

See *gamma radiation*.

geometric mean

The *n*th root of the product of all the members of a set of positive numbers, where *n* is the number of members.

geometric standard deviation (GSD)

In probability theory and statistics, the spread of a set of numbers whose preferred average is the geometric mean.

half-life

Time in which half of a given quantity of a particular radionuclide disintegrates (decays) into another nuclear form. During one half-life, the number of atoms of a particular radionuclide decreases by one half. Each radionuclide has a unique half-life ranging from millionths of a second to billions of years.

intake

Radioactive material taken into the body by inhalation, absorption through the skin, injection, ingestion, or through wounds.

ionizing radiation

Radiation of high enough energy to remove an electron from a struck atom and leave behind a positively charged ion. High enough doses of ionizing radiation can cause cellular damage. Ionizing particles include alpha particles, beta particles, gamma rays, X-rays, neutrons, high-

speed electrons, high-speed protons, photoelectrons, Compton electrons, positron/negatron pairs from photon radiation, and scattered nuclei from fast neutrons. See *alpha radiation*, *beta radiation*, *gamma radiation*, *neutron radiation*, *photon radiation*, and *X-ray radiation*.

isotope

One of two or more atoms of a particular element that have the same number of protons (atomic number) but different numbers of neutrons in their nuclei (e.g., ^{234}U, ^{235}U, and ^{238}U). Isotopes have very nearly the same chemical properties.

monitoring

Periodic or continuous determination of the presence or amount of ionizing radiation or radioactive contamination in air, surface water, groundwater, soil, sediment, equipment surfaces, or personnel (for example, bioassay or alpha scans). In relation to personnel, monitoring includes internal and external dosimetry including interpretation of the measurements.

National Council on Radiation Protection and Measurements (NCRP)

Private U.S. public service organization chartered by the U.S. Congress to formulate and disseminate information, guidance, and recommendations on radiation protection and measurements.

National Institute for Occupational Safety and Health (NIOSH)

U.S. agency responsible for dose reconstruction under the Energy Employees Occupational Illness Compensation Program. Part of the Centers for Disease Control and Prevention, which is part of the U.S. Department of Health and Human Services, NIOSH is the Federal agency responsible for conducting research and making recommendations for the prevention of work-related injury and illness.

neutron (n)

Basic nucleic particle that is electrically neutral with mass slightly greater than that of a proton. There are neutrons in the nuclei of every atom heavier than normal hydrogen. See *element*.

neutron radiation

Radiation that consists of free neutrons unattached to other subatomic particles emitted from a decaying radionuclide. Neutron radiation can cause further fission in fissionable material such as the chain reactions in nuclear reactors, and nonradioactive nuclides can become radioactive by absorbing free neutrons. See *neutron*.

noble gas

Gaseous chemical element that does not easily react chemically with other elements (i.e., they are inert).

nuclide

Stable or unstable isotope of any element. Nuclide relates to the atomic mass, which is the sum of the number of protons and neutrons in the nucleus of an atom. A radionuclide is an unstable nuclide.

photon

Quantum of electromagnetic energy generally regarded as a discrete particle having zero rest mass, no electric charge, and an indefinitely long lifetime. The entire range of electromagnetic radiation that extends in frequency from 10^{23} cycles per second (hertz) to 0 hertz.

photon radiation
Electromagnetic radiation that consists of quanta of energy (photons) from radiofrequency waves to gamma rays.

probability of causation (POC)
For purposes of dose reconstruction for the Energy Employees Occupational Illness Compensation Act, the percent likelihood, at the 99th percentile, that a worker incurred a particular cancer from occupational exposure to radiation.

radiation
As used in this document, *ionizing radiation*.

radiation monitoring
See *monitoring*.

radiation field
Area around a source into which radiation dissipates.

radiation source
(1) Any object or substance that emits radiation. (2) Package of radioactive material constructed to have specific radiation properties used for example for medical purposes or to calibrate dosimeters.

radiation type
Of importance in health physics, alpha, beta, photon (X- or gamma), or neutron radiation.

radioactive
Of, caused by, or exhibiting radioactivity.

radioactivity
Property possessed by some elements (e.g., uranium) or isotopes (e.g., ^{14}C) of spontaneously emitting energetic particles (electrons or alpha particles) by the disintegration of their atomic nuclei.

radionuclide
Radioactive nuclide. See *radioactive* and *nuclide*.

reactor
Device in which a fission chain reaction occurs under controlled conditions to produce heat or useful radiation for experimental purposes or to generate electrical power or nuclear fuel.

reactor vessel
Structure enclosing the fuel elements, control elements, coolant piping, and other structures that support the reactor core.

rem
Traditional unit of radiation dose equivalent that indicates the biological damage caused by radiation equivalent to that caused by 1 rad of high-penetration X-rays multiplied by a quality factor. The sievert is the International System unit; 1 rem equals 0.01 sievert. The word derives from roentgen equivalent in man; rem is also the plural.

roentgen (R)

Unit of photon (gamma or X-ray) exposure for which the resultant ionization liberates a positive or negative charge equal to 2.58×10^{-4} coulombs per kilogram (or 1 electrostatic unit of electricity per cubic centimeter) of dry air at 0°C and standard atmospheric pressure. An exposure of 1 R is approximately equivalent to an absorbed dose of 1 rad in soft tissue for higher energy photons (generally greater than 100 kiloelectron-volts).

routine monitoring

Monitoring carried out at regular intervals during normal operations.

shallow dose equivalent

Dose equivalent in units of rem or sievert at a depth of 0.07 millimeters (7 milligrams per square centimeter) in tissue equal to the sum of the penetrating and nonpenetrating doses.

sievert

International System unit for dose equivalent, which indicates the biological damage caused by radiation. The unit is the radiation value in gray (equal to 1 joule per kilogram) multiplied by a weighting factor for the type of radiation and a weighting factor for the tissue; 1 Sv equals 100 rem.

solubility type

See *absorption type*.

technical basis document (TBD)

Oak Ridge Associated Universities Team document that describes a feature of a site in relation to the Energy Employees Occupational Illness Compensation Program. See *site profile*.

thermoluminescence

Property that causes a material to emit light as a result of heat.

thermoluminescent dosimeter (TLD)

Device for measuring radiation dose that consists of a holder containing solid chips of material that, when heated by radiation, release the stored energy as light. The measurement of this light provides a measurement of absorbed dose.

thermoluminescent dosimeter chip

Small block or crystal of lithium fluoride in a thermoluminescent dosimeter. A TLD-600 dosimeter contains a chip made from more than 95% ^{6}Li for neutron radiation detection, and a TLD-700 dosimeter contains a chip made from more than 99.9% ^{7}Li for photon and beta radiation detection. Also called crystals.

U.S. Atomic Energy Commission (AEC)

Federal agency created in 1946 to assume the responsibilities of the Manhattan Engineer District (nuclear weapons) and to manage the development, use, and control of nuclear energy for military and civilian applications. The U.S. Energy Research and Development Administration and the U.S. Nuclear Regulatory Commission assumed separate duties from the AEC in 1974. The U.S. Department of Energy succeeded the U.S. Energy Research and Development Administration in 1979.

U.S. Department of Energy (DOE)

Federal agency created in 1979 that assumed, from the Energy and Research Development Administration, the responsibility for development of new reactors along with production and use of nuclear materials and weapons by the Federal government.

U.S. Department of Labor (DOL)

Agency that oversees compliance with Federal labor laws as well as collects labor-related information. DOL is responsible for compensation decisions under the Energy Employees Occupational Illness Compensation Program Act.

U.S. Energy Research and Development Administration (ERDA)

One of the successors to the U.S. Atomic Energy Commission, the duties of which ERDA and the U.S. Nuclear Regulatory Commission assumed in 1974. ERDA assumed the responsibility for development of new reactors along with production and use of nuclear materials by the Federal government, and the U.S. Nuclear Regulatory Commission assumed responsibility for oversight of the commercial nuclear power industry. The U.S. Department of Energy succeeded ERDA in 1979.

X-ray radiation

Electromagnetic radiation (photons) produced by bombardment of atoms by accelerated particles. X-rays are produced by various mechanisms including bremsstrahlung and electron shell transitions within atoms (characteristic X-rays). Once formed, there is no difference between X-rays and gamma rays, but gamma photons originate inside the nucleus of an atom.