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DRAFT REPORT

Identification and Prioritization of Radionuclide Releases from the Idaho National Engineering and Environmental Laboratory

Centers for Disease Control and Prevention

Department of Health and Human Services

September 30, 2000

Submitted to the Center for Disease Control and Prevention
in Partial Fulfillment of Contract No. 200-95-0927

"Setting the standard in environmental health"

RAC

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EXECUTIVE SUMMARY

The objective of the current work under Task Order 5 for the Centers for Disease Control and Prevention (CDC) has been to use screening methods to prioritize past releases of radioactive materials to the environment from the Idaho National Engineering and Environmental Laboratory (INEEL) for the period 1952–1992. This work involved identifying the facilities and programs that were the source of important releases and time periods when they occurred. We also considered the particular radioactive materials released, the quantities released from the INEEL Site, and their potential for contributing to dose at locations where members of the public might be exposed. Activities at the INEEL have resulted in routine operational releases that took place throughout the year and isolated episodic releases that took place only during a period of days or weeks.

We used screening methods to prioritize the radioactive materials (radionuclides) released to the environment in terms of their importance to human health. The result is reported as a screening value. The screening value is reported in units of dose (millirem or sievert). This value does not represent a "true dose" because the many conservative assumptions used in the screening analysis for routine and episodic releases tend to maximize the estimates of potential dose to a hypothetical person. It is reasonable to assume that the true dose under more realistic situations would be lower than the screening dose or value.

We evaluated the routine operational releases separately from the episodic or short-term releases because of the different assumptions and methods used to assess their potential importance. For routine operational releases, we reviewed airborne, liquid, and solid radioactive waste disposal procedures and discharge information at the Site to determine potential exposure pathways for members of the public from these releases. Liquid wastes were discharged to disposal wells, seepage basins or pits, or seepage ponds depending upon the facility, and they were generally monitored at the time of release. Surface streams or rivers do not flow from the INEEL Site to offsite locations. The INEEL has used several areas for solid radioactive waste materials disposal. Groundwater serves as a potential exposure pathway from liquid effluent disposal areas and leakage from underground storage of solid waste at the radioactive waste areas at the INEEL. We evaluated groundwater as a potential historic exposure pathway. A number of radioactive contaminants, including tritium, ^{90}Sr, ^{60}Co, ^{137}Cs, ^{129}I, ^{238}Pu, $^{239,240}Pu$, and ^{241}Am, have been detected in groundwater. These contaminants have been measured by the United States Geological Survey, and those data were used in this analysis. A screening analysis was done only for tritium in groundwater because it had been detected in wells at the boundary during the years 1983, 1984, and 1985, and it is the only radionuclide that has been detected at Site boundary locations. This analysis resulted in a screening value of 0.06 mrem (0.0000006 Sv), which relates only to offsite exposures to groundwater in the past and does not make any judgments regarding onsite exposures or future offsite exposures to other radionuclides.

Radionuclides released routinely to air from the INEEL were primarily those expected from reactor and reprocessing operations. Large amounts of the release data for the early years of plant operation, when releases were generally greatest, were in the form of gross activity. The release data measured since the mid-1970s were of good quality and may be sufficient for dose reconstruction to determine quantities of important

individual radionuclides. We used information from a variety of sources but focused primarily on Site documents reporting effluent monitoring procedures, changes, and data. Because the majority of historic monitoring and record-keeping came from the Site, we relied on available Site records for our screening efforts. We compared and confirmed information and data in summary reports with original or supporting data from daily, weekly, or monthly reports for a select period of time to ensure that the summary documents were accurate.

In addition to Site monitoring and process records, we also drew on the basic chemistry and nuclear physics of the reactor and chemical plant operations at the INEEL. Based on this type of information, it was possible to estimate the types and relative quantities of materials that might be expected from a particular process or reactor operation run. This information was useful for radionuclides that were not measured during particular years of operations, or from particular facilities. We prioritized the key radionuclides released to air from the INEEL facilities as potential contributors to offsite radiation dose, reported as the screening value, using the screening methods developed by the National Council on Radiation Protection and Measurements (NCRP).

For routine releases from the INEEL, we evaluated all pathways of exposure in the NCRP methodology and all individual years for 62 radionuclides released between 1952 and 1992. We assumed two different locations for the routine screening analysis: (1) an offsite location at Atomic City where all pathways of exposure were considered (the point of maximum offsite concentration where someone lived), and (2) an onsite location near Highway 20, where the inhalation and plume immersion pathways would be most important for exposure. Radionuclides released to air from the INEEL were evaluated or screened to determine the radionuclides, time periods, and facilities that contributed most significantly to the screening values at these locations.

The output from this screening was a list of the radionuclides and their relative contribution to the screening values from all pathways (or the inhalation and plume immersion pathways) of exposure to the nearby individual. We compiled the release estimates and results of the NCRP screening methodology in Excel spreadsheets. For radionuclides released to air from the INEEL, the key contributors to the screening value at the offsite location, when all pathways were considered, were ^{137}Cs, ^{131}I, and ^{90}Sr, whether we consider all years of release together or focus on the high release years during the late 1950s. However, it is clear that releases during the Radioactive Lanthanum (RaLa) program, which occurred during the high release years, shifted the focus of the most important radionuclide from ^{137}Cs to ^{131}I.

At the onsite location, where the inhalation and plume immersion pathways were most important, a greater variety of radionuclides contributed to the screening value. The key contributors to the screening value for most of the early years included ^{41}Ar, some krypton and xenon isotopes, ^{144}Ce, ^{90}Sr, ^{106}Ru, and ^{131}I. However, when screening results for an individual year were considered, other important radionuclides may have been major contributors to the screening value. For example in 1964, the year with the highest screening value at the onsite location, ^{106}Ru emerged as a key contributor to the screening value along with ^{90}Sr, ^{137}Cs, and ^{41}Ar.

Releases from the Idaho Chemical Processing Plant (ICPP) were the primary contributors to the screening value during most early operational years after 1953. The Test Reactor Area releases were the primary contributors to the screening value during 1952 and 1953, the first 2 years of plant operation.

The INEEL had many episodic events that resulted in the release of unknown quantities of many different radionuclides. These events were shorter term releases and included planned research programs such as the Initial Engine Tests, the RaLa program, and the Fission Product Field Release Tests, and accidents such as a criticality that occurred at the ICPP in 1959 and the SL-1 reactor accident in 1961. For these events, it was necessary to first reconstruct the episodic event and estimate the radionuclides that were likely present during the release and then estimate the fraction of those radionuclides present that was actually released to the environment. For the episodic events, we evaluated the potential for exposure by applying atmospheric screening factors to the total amount or quantity of individual radionuclides released for a given episodic release because the precise duration of each release was not known in all cases. This approach resulted in a calculated total integrated concentration at the location of exposure, which also simplified the calculation, allowed for a consistent method of screening and comparing each release event, and helped minimize the bias created by evaluating a calculated episodic concentration using screening factors that assume an annual average concentration.

We evaluated and calculated relative screening values for a total of 134 individual episodic release events. By comparison, the U.S. Department of Energy Historic Dose Evaluation report evaluated and calculated doses for a total of 54 episodic release events, including several Initial Engine Test runs that were divided into more than one time period. Our simplified screening methodology allowed for a more comprehensive approach to evaluating each episodic release event for which it was possible to construct a source term. We attempted to evaluate all of the episodic releases in a consistent, conservative manner. Where specific information was unavailable for a given release, we made conservative assumptions. Because of the difficulty of comparing releases of varying duration and often discontinuous nature, we divided the releases into four separate exposure categories: (1) single-day release, onsite exposure, (2) single-day release, offsite exposure, (3) multi-day release, onsite exposure, and (4) multi-day release, offsite exposure. The release events that had high relative screening values in both onsite and offsite categories were the

- Fuel Element Burn Test -B in 1957
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We evaluated a special exposure scenario concerning duck hunters for the potential exposure pathways of ingestion, external irradiation from ^{137}Cs contamination, and inhalation of airborne ^{137}Cs and $^{239,240}Pu$ contamination associated with plucking and using feathers in constructing a pillow. This report provides ingestion dose calculations for ^{137}Cs, ^{134}Cs, ^{75}Se, ^{131}I, and $^{239,240}Pu$ contamination of muscle and liver, using average and maximum measured concentrations in

ducks from the Test Reactor Area Ponds. Ingesting the duck meat was the most important exposure pathway and ^{137}Cs was the largest contributor to the ingestion dose. Based on these very conservative assumptions, we calculated the hunter's dose was 12 mrem (or 64 mrem if the maximum ^{137}Cs concentration is used in the calculations) from eating one contaminated duck with the average measured radionuclide concentration. This may be an important exposure pathway for some individuals in the INEEL region.

Our screening analysis work identified some potential areas of consideration if additional resources and time were to be focused at the INEEL. We concluded that several episodic release events contributed most to the potential exposure of members of the public. Routine releases from the ICPP, especially for the 1957 through 1963 period, were also important for the INEEL region. A future detailed study of ICPP releases for 1 or more years during this time (e.g., 1956, 1957, 1958, or 1959) would allow for an assessment of health impacts associated with both routine annual releases and with shorter term episodic releases. The relative importance of all INEEL releases could be determined if this future study were combined with a detailed investigation of a few other important episodic releases identified in Task Order 5, particularly during the early years. This detailed investigation could also address additional issues, such as the potential importance of short-lived radionuclides, release fraction uncertainties, and onsite exposures. A complete evaluation of Site-wide releases for 1 or more years would enable researchers to assess health impacts related to routine releases (focused on ICPP releases) and episodic releases occurring. A number of the highest-ranking episodic releases occurred during 1957 and 1958, suggesting those years as a practical focus for additional research at the INEEL.

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TABLES

ACRONYMS

AEC	U.S. Atomic Energy Commission
ANL-W	Argonne National Laboratory-West
ANP	Aircraft Nuclear Propulsion (Program)
APS	Atmospheric Protection System
ARA	Auxiliary Reactor Area
ATR	Advanced Test Reactor
BOOT	Burn Out One Tube
BORAX	Boiling Water Reactor Experiment
CDC	Centers for Disease Control and Prevention
CERT	Controlled Environmental Radioiodine Test
CERT	Controlled Environmental Release Test
CFA	Central Facilities Area
CFSGF	Coal-Fired Steam Generating Facility
CTF	Core Test Facility
DOE	U.S. Department of Energy
EBR-I	Experimental Breeder Reactor No. 1
EBR-II	Experimental Breeder Reactor No. 2
ECF	Expended Core Facility
EFS	Experimental Field Station
EPA	U.S. Environmental Protection Agency
ERDA	Energy Research and Development Administration (replaced by DOE)
ETR	Engineering Test Reactor
EXCES	Experimental Cloud Exposure Study
FAST	Fluorinel Dissolution Process and Fuel Storage Facility
FECF	Fuel Element Cutting Facility
FEET	Fuel Element Effluent Test
FEBT	Fuel Element Burn Test
FPFRT	Fission Product Field Release Test
FPR	fission product radio
GRID III	Test area where the Fuel Element Burn Tests A and B were conducted
GSF	Graphite Storage Facility
HEPA	high-efficiency particulate air (filter)
HES	(INEEL) Health Effects Subcommittee
HFEF	Hot Fuel Examination Facility
HTRE	Heat Transfer Reactor Experiment
ICPP	Idaho Chemical Processing Plant

IET	Initial Engine Test
ILTSF	Intermediate Level Transuranic Storage Facility
INEEL	Idaho National Engineering and Environmental Laboratory
LDDT	Long Distance Diffusion Text
LIME	Limited Melt Experiment
LOFT	Loss of Fluid Test
MTR	Materials Testing Reactor
NCRP	National Council on Radiation Protection and Measurements
NRF	Naval Reactors Facility
NRTS	National Reactor Testing Station, original name for the Site (1949-1974)
NWCF	New Waste Calcining Facility
ORME	Organic Moderated Reactor Experiment
PBF	Power Burst Facility
RAC	*Risk Assessment Corporation*
RAL	Remote Analytical Laboratory
RaLa	Radioactive Lanthanum
RSAC-4	Radiological Safety Analysis Computer Program, Version 4
RWMC	Radioactive Waste Management Complex
RWMIS	Radioactive Waste Management Information System
SL-1	Stationary Low-Power Reactor No. 1
SNAPTRAN	SNAP 10A Transient
SPERT	Special Power Excursion Reactor Test
TAN	Test Area North
TDA	Transuranic Disposal Area
TRA	Test Reactor Area
TREAT	Transient Reactor Test Facility
TSA	Transuranic Storage Area
TSF	Technical Services Facility
USGS	U.S. Geological Survey
UTM	Universal Transverse Mercator
WCF	Waste Calcining Facility
WERF	Waste Experimental Reduction Facility
ZPPR	Zero Power Plutonium Reactor (or, later on, Zero Power Physics Reactor)

INTRODUCTION

The objective of Task Order 5 for the Center for Disease Control and Prevention (CDC) has been to use screening methods to evaluate important releases of radioactive materials to the environment from the Idaho National Engineering and Environmental Laboratory (INEEL). This work involves identifying the facilities and programs that were the sources of important releases considering the particular radionuclides released, the quantity released from the site, and the potential for contributing to dose at locations where members of the public might be exposed. These objectives have been accomplished in a step-wise fashion by ranking or prioritizing those facilities or special programs and operations whose releases may warrant further study. Reconstructing releases of radioactive materials from a facility like the INEEL can be a time-consuming and resource-intensive process. Reconstructing historic releases from a site like the INEEL should be a staged process involving key steps to identify where more detailed work may be required. One of the first, critical steps in such a study is using conservative[1] screening methods to focus later work on the most important contaminants released from the facility to the environment.

Screening is a method for prioritizing the radioactive materials (radionuclides) released to the environment in terms of their importance to human health. Because the INEEL released numerous materials to the environment, screening calculations provide an objective basis for making decisions about priorities for further study and for allowing researchers to focus on the radionuclides, facilities, programs, and time periods that are most important in terms of exposing individuals or local populations. At the INEEL, radionuclides were released on a routine, long-term basis from the facilities carrying out their normal activities. Episodic or short-term releases also occurred during accidents, unplanned events, and planned releases tests, many from unmonitored sources.

We evaluated the routine operational releases separately from the episodic releases because of the different assumptions and methodology used to assess their potential importance. For routine operational releases, we ranked the key radionuclides released to air or water from the INEEL facilities as potential contributors to offsite radiation dose based on a screening value[2] calculated with the screening methods developed by the National Council on Radiation Protection and Measurements (NCRP) (NCRP 1996). This methodology provides an understandable and conservative way to prioritize radionuclides released over the long term so that decisions can be focused on the most important contributors to doses to members of the public.

For the episodic events, the potential for exposure was evaluated by applying the atmospheric screening factors developed by the National Council on Radiation Protection and Measurements (NCRP 1996) to the entire quantity of individual radionuclides released during a

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[1] By conservative, we mean that we have selected parameter values (e.g., gross activity and fractional release estimates), used dispersion calculation methodology, and assumed exposure locations for each calculation that forced our estimated screening values to likely be significantly higher than actual dose estimates to ensure that we have not underestimated the potential impact of any release.

[2] The screening value is reported in units of dose (millirem or sievert), but does not represent a "true dose" because of many conservative assumptions used in the screening analysis for both routine and episodic releases

given event. Several episodic events resulted in the release of known quantities of specific radionuclides to the atmosphere, and these are relatively simple to evaluate in terms of potential dose to exposed individuals using the atmospheric screening factors.

Other episodic events resulted in the release of unknown quantities of many different radionuclides to the atmosphere. For these events, it was necessary to first reconstruct the episodic event and estimate the radionuclides that were likely present during the release and then estimate the fraction of those radionuclides present that was actually released to the environment. Once this was completed, we used the NCRP (1996) screening factors to evaluate the relative important of each release.

BACKGROUND

In January 1990 the Secretaries of the U.S. Departments of Energy (DOE) and the U.S. Department of Health and Human Services signed a Memorandum of Understanding transferring responsibility and funding for studies of chemical and radionuclide releases from DOE nuclear facilities, and the potential exposures and health effects to the surrounding population to the U.S. Department of Health and Human Services. In August 1991, DOE published a historical radiation dose evaluation for the INEEL. A DOE review committee recommended a more detailed study using source documents and incorporating public involvement; and the Governor of Idaho asked the CDC to perform this study. CDC established the INEEL Health Effects Subcommittee (HES), a Federal advisory committee, whose members provide advice to CDC on community concerns about CDC's activities with regard to dose reconstruction at INEEL. The CDC completed the first phase of a dose reconstruction study, developing a database of documents at the INEEL relevant to an environmental dose reconstruction, in 1994 (SC&A 1994).

In 1997, the CDC funded *Risk Assessment Corporation (RAC)* to complete Task Orders 1 and 5 to review the Phase I database of documents, and any others that may be relevant, to catalog INEEL releases of chemicals and radionuclides, screen these releases, and list in order by priority those releases that may warrant further study. In September 1999, *RAC* completed Task Order 1, which determined the feasibility of estimating doses to members of the public from toxic chemicals released in the past from the INEEL. That report concluded that the evaluated chemicals had not been released in quantities sufficient to warrant a dose reconstruction and assessment of past health risk offsite (McGavran and Case 1999).

To complete the Task Order 5 work, *RAC* used the Phase I database as a starting point for our information sources. We searched the Phase I database for relevant documents using accepted search methods (e.g., keywords), and talked to personnel involved in past site operations and effluent monitoring. We retrieved and copied relevant documents from the existing Phase I database that are important for understanding the major release points and radionuclides released historically from operations at the INEEL. As work on Task Order 5 proceeded, it became clear that documents in a number of the boxes at the INEEL had not been entered into the Phase I database as individual items, but rather as part of the entire box. In addition, many documents entered in the Phase I database were not photocopied, and some documents could not be tracked because they had been moved from the original INEEL or offsite location recorded in the Phase I database.

As a result, Task Order 6 was initiated by the INEEL HES and approved by CDC to retrieve, review, and copy relevant documents that may not have been completely evaluated during Phase

I. It was thought that these documents might be important in completing the initial screening activities for radioactive materials and for chemicals. The Task Order 6 work resulted in developing the document database that incorporated many of the Phase I records along with the additional records reviewed. Task Order 6 was completed and the INEEL Task Order database was submitted to CDC for review in September 2000. It has not been a part of the scope of this task to change the structure and function of the original Phase I database.

As a part of Task Order 5, we prioritized historical releases of radionuclides from the INEEL so that greatest attention can be focused on those releases that appear most significant in terms of health effects for those living offsite or those periodically onsite as part of their occupation. Task Order 5 is different from Task Order 1, the chemical feasibility study, in several ways. In general, there is more measurement information and monitoring data for radioactive materials, or radionuclides, that were released. In addition, various studies have evaluated particular release events, and government programs at the INEEL.

The INEEL is unique among the DOE facilities because it is a large complex site with many independent contractors, goals, and missions. Our approach to Task Order 5 was to examine the site as a whole with respect to total radionuclide releases and then to focus more attention on the facilities, years or programs that appear to have been the largest contributors to releases.

REVIEW OF THE SITE FACILITIES AND ACTIVITIES

In this section a brief overview of the history and layout of INEEL is provided. The total reported radionuclide releases from the site to air and via liquid effluent are reviewed separately to identify the time periods during which the largest releases occurred, and the facilities and processes that contributed the most to these releases. These facilities are then discussed in turn with regard to their function and purpose, and the air and liquid effluent monitoring systems that were in place. We discuss our screening approaches for the radionuclide releases from routine operations and from the episodic events separately.

In 1949, the Atomic Energy Commission (AEC) established the National Reactor Testing Station (NRTS) as a government reserve for building, testing, and operating nuclear reactors, support facilities and equipment, and associated activities and program. The site, situated on the Snake River Plain of southeastern Idaho at an elevation of about 5000 ft., is located directly above the Snake River Plain Aquifer. The site encompasses almost 572,000 acres with a maximum distance of about 39 miles from north to south and 36 miles from east to west at the southern boundary. In 1974, the NRTS was renamed the Idaho National Engineering Laboratory (INEL), and in 1997 was designated the Idaho National Engineering and Environmental Laboratory (INEEL). Although this current report is an assessment of historical releases, the current INEEL name is used throughout this report. The population within a 50-mile radius of the site was approximately 70,000 in 1970 (ERDA 1977) and about 120,000 people in 1990 (Hoff et al. 1992). The INEEL has operated 52 reactors plus fuel handling and reprocessing facilities and radioactive waste storage facilities since it began operations. By 1988, 13 reactors were still "operating or operable." Figure 1 [3] shows the major facilities and their locations at the INEEL.

[3] The underlined figure and table references and the citations in the printed report indicate hyper linking to the referenced figure, table, or reference in the electronic version of the report.

Activities at the INEEL have resulted in chronic operational releases taking place throughout the year as well as isolated episodic releases that may have taken place only during a period of days or weeks. General information on these specific facilities and operations can be found in several sources (ERDA 1977; Bowman et al. 1984; DOE 1991a; SC&A 1994; McGavran and Case 1999; Litteer and Reagan 1989). It is important to evaluate these routine operational releases separately from episodic releases to assist with decisions about the need for future dose reconstruction efforts. Most of the episodic releases with the potential to create an appreciable offsite dose occurred during the late 1950s and early 1960s, and, in many cases, radionuclide releases must be reconstructed. This is also true for many of the operational releases occurring during this time period.

The radionuclide release data measured since the mid-1970s is good quality and may be sufficient for dose reconstruction purposes with regard to quantities of specific important radionuclides. Before that time, particularly before 1962, much of the reported release data are in the form of gross activity only, which is inadequate for radionuclide screening purposes. Additional sources of reported effluent monitoring data are available in weekly and monthly reports, particularly from the Idaho Chemical Processing Plant (ICPP) (Hayden 1957-1963; Rich 1962; Williamson 1977). Scientists conducted an historical dose evaluation in 1991 and identified apparent technical errors with some of the release data (DOE 1991a). These factors necessitated a careful review of reported and recalculated releases to estimate individual radionuclide content for the purpose of screening and establishing the relative importance of releases. Because of gradual improvements in monitoring techniques through time, our evaluation of reported releases is first directed at those years for which release data are most uncertain. Table A1 in Appendix A provides a summary of major release points for airborne, liquid, and solid radioactive wastes at the INEEL in place in the 1970s. There is additional information available on the potential sources of radioactive airborne effluents from ICPP. TRA, CFA, ARA, the SPERT-PBF Reactor Area, ANL-W, and NRF (Hogg et al. 1971a).

Responsibilities for various effluent measurement programs at the INEEL were borne by the individual facilities. By the 1980s, the Radiological and Environmental Sciences Laboratory had overall responsibility for monitoring outside the contractor facilities. The United States Geological Survey (USGS) and the National Oceanic and Atmospheric Administration perform groundwater and meteorological monitoring, respectively. Figure 2 provides a perspective on total releases to air from the INEEL facilities.

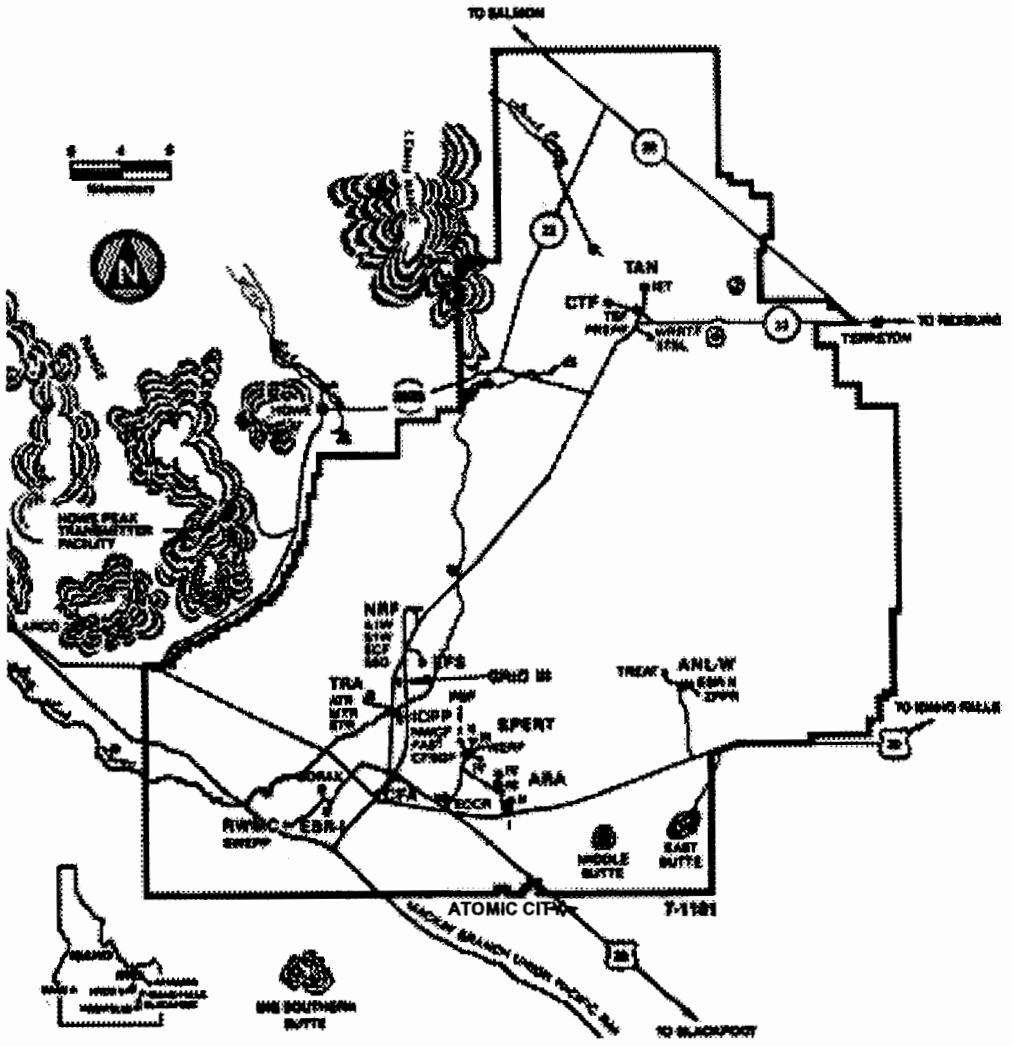

Figure 1. Map showing locations of various facilities and projects at the Idaho National Engineering and Environmental Laboratory. These facilities include the Argonne National Laboratory-West (ANL-W) where the Experimental Breeder Reactor No. 2 (EBR-II), Transient Reactor Test Facility (TREAT) and Zero Power Plutonium Reactor (ZPPR)[4] are located, Auxiliary Reactor Area (ARA), Central Facilities Area (CFA), Idaho Chemical Processing Plant (ICPP), Experimental Breeder Reactor No. 1 (EBR-I), Boiling Water Reactor Experiment (BORAX), Radioactive Waste Management Complex (RWMC), Special Power Excursion Reactor Test (SPERT) area, GRID III, the test grid where the Fuel Element Burn Tests A and B were conducted, Test Reactor Area (TRA), the Experimental Field Station (EFS), Naval Reactors Facility (NRF), Test Area North (TAN) where the Initial Engine Tests (IETs) were conducted, and the Core Test Facility (CTF) at TAN. The Big and Little Lost

[4] Later changed to Zero Power Physics Reactor.

Rivers and Birch Creek originate in the mountains to the northwest and flow toward the INEEL. The surface streams and rivers do not flow from onsite to offsite locations.

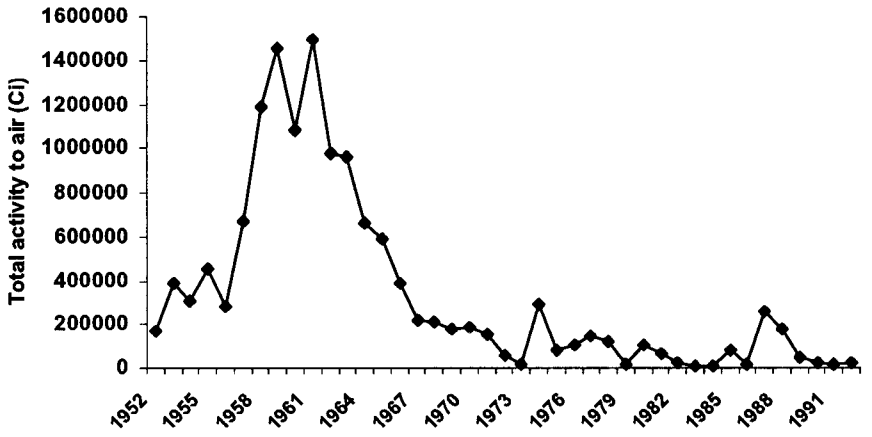

Figure 2. Reported total releases of radioactivity to air from facilities at the INEEL (DOE 1991a; Hoff et al. 1992; Hoff et al. 1993).

Figure 2 shows that total releases were highest from 1957 through the mid-1960s when over 500,000 curies (Ci) was released annually. In the 1970s, the annual average release was 116,000 Ci. This average annual release dropped to about 80,000 Ci in the 1980s and to 21,000 Ci for 1990 through 1992. While total radioactivity cannot be used for screening purposes, this temporal distribution of releases provides a perspective for estimating releases from various facilities and areas of operation onsite. Between 1952 and 1974, reported releases of total activity from the site were 12 million Ci (DOE 1991a; ERDA 1977). Figure 3 shows that the releases of radioactivity to air were much higher for some facilities than others, and that the Idaho Chemical Processing Plant (ICPP) and the Test Reactor Area (TRA) were the largest contributors to atmospheric releases from the INEEL. The ICPP contributed about 8 million Ci, and the TRA about 5.5 million Ci to the total reported site releases from the early 1953 through the late 1980s. In comparison, the Test Area North and Argonne National Laboratory-West contributed only about 30,000 Ci each.

Liquid wastes were discharged to injection wells[5], seepage basins or pits, or seepage ponds depending upon the facility, and were generally monitored at the time of release. Figure 4 shows that, as with the airborne releases, the ICPP and TRA areas have discharged greater than 95% of the total liquid radioactivity at the INEEL for all years. The INEEL site has no surface streams or rivers flowing from onsite to offsite locations.

[5] Injection wells were drilled for the disposal of liquid wastes from the INEEL and were not used for drinking water or other purposes.

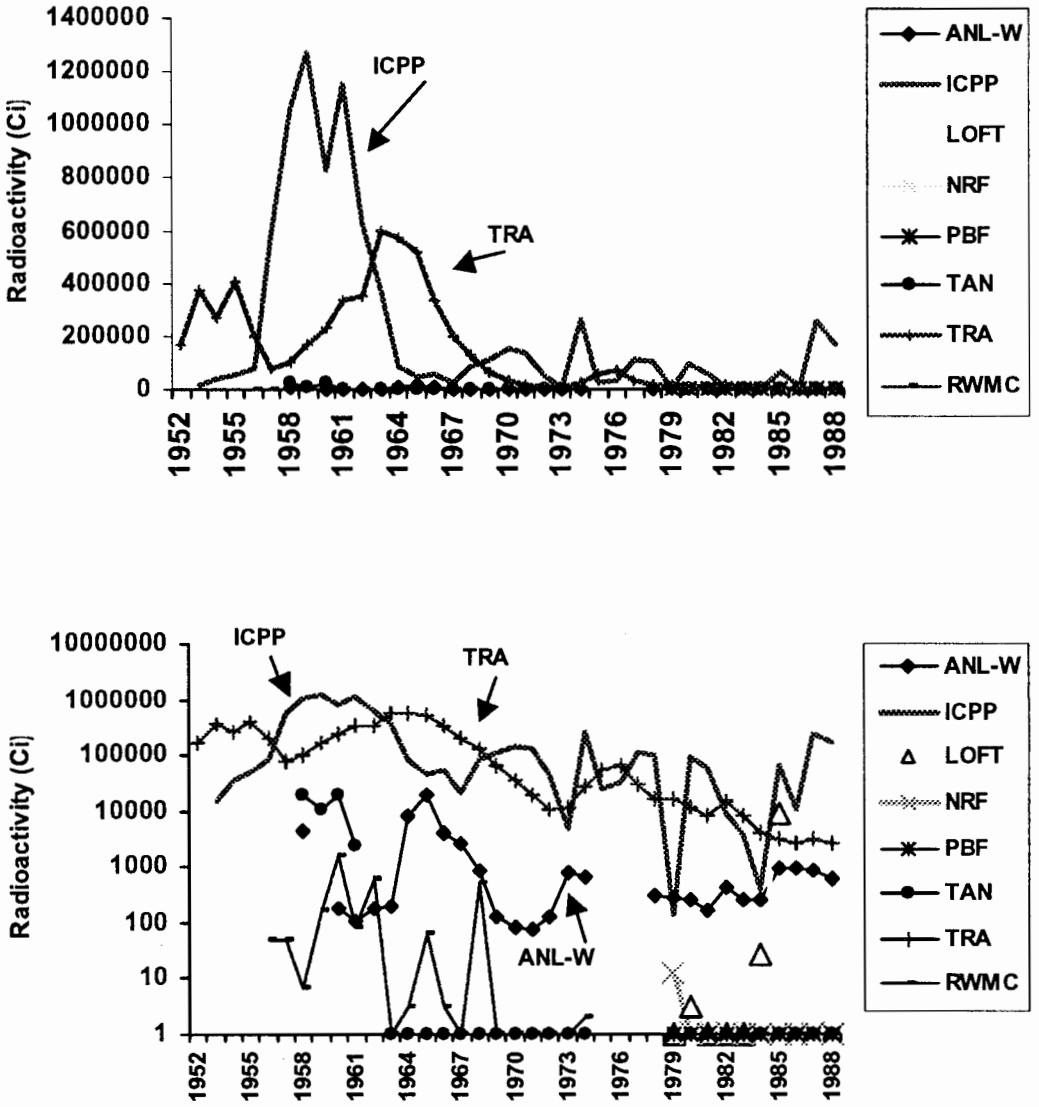

Figure 3. Reported releases of radioactivity to air from 1952 through 1988 for various facilities and programs onsite (DOE 1991a). The top graphic shows the facility release data on a linear scale and the bottom graphic shows the same release data on a logarithmic scale.

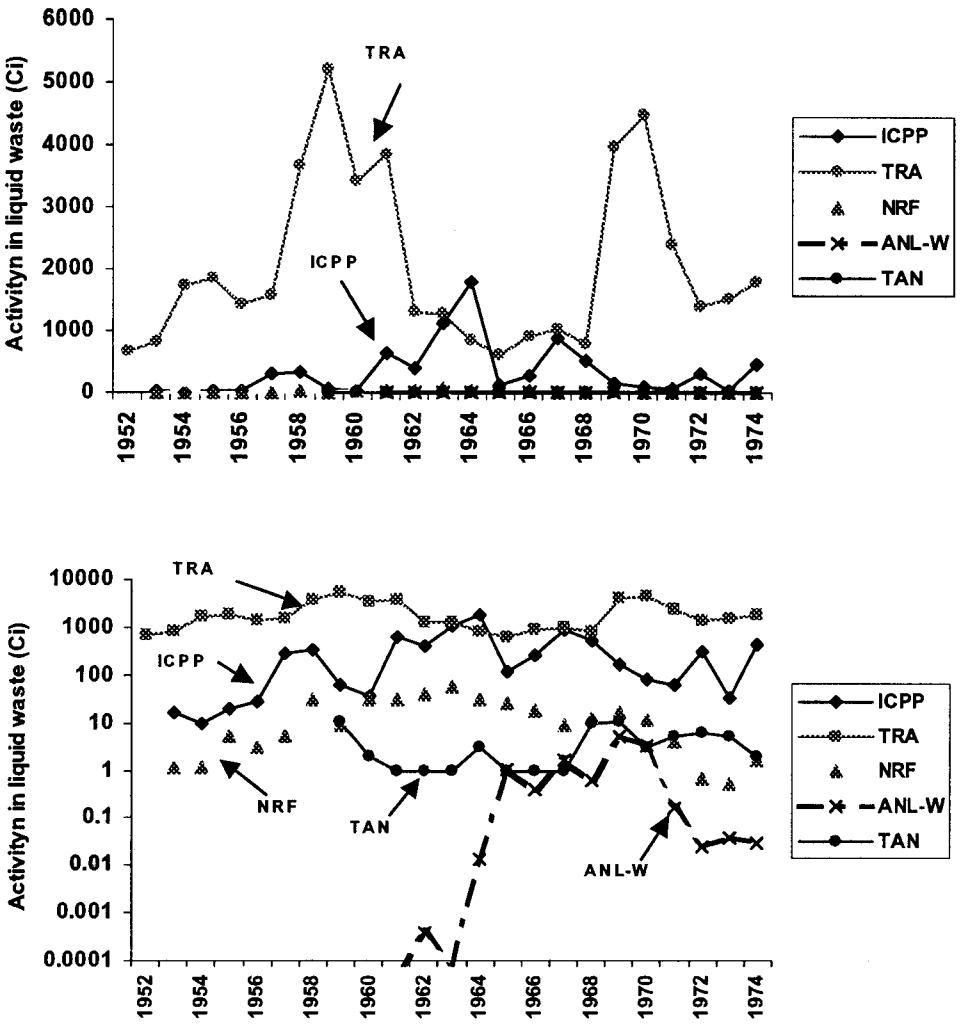

Figure 4. Releases of radioactivity in liquid wastes from five facilities at the INEEL. The Test Reactor Area was the largest contributor to activity in liquid effluents. For releases of liquid wastes in the early years, discharges were made to wells, seepage basins or pits, or seepage ponds depending upon the facility. The top graphic shows releases on a linear scale and the bottom graphic displays the same release data on a logarithmic scale.

Test Reactor Area

The TRA (Figure 5) is a complex containing support facilities and three test reactors: the Materials Testing Reactor (MTR), the Engineering Test Reactor (ETR), and the Advanced Test Reactor (ATR). At TRA, scientists studied the performance of reactor materials and equipment components under high neutron flux conditions. The ventilation system was designed for the reactors to use outside air, first brought through office areas, then to slightly contaminated areas, and finally into high radiation areas. From these high radiation areas, the air was filtered an

ddischarged to the stack. Negative pressure was maintined in contaminated and high radation areas (Hogg et al. 1971b). MTR began operations on March 31, 1952, at power level of 30 MW(t), and it provided fuel for a special program, the Radioactive Lanthanum (RaLa) Program, carried out from 1957 through 1963 at ICPP. The ETR startup occurred in 1957 with operating power level of 175 MW(t), and the ATR was the world's largest test reactor when it began operations in July 1967. The TRA had three 250-foot stacks with monitors for gross alpha and beta, a gamma detector, and a charcoal filter for iodine collection. Radioactive airborne effluents were discharged through three stacks from the ATR, ETR, and MTR. Descriptions of the gaseous waste systems and monitorinf setups are available (Hogg et al. 1971b). From 1952–1974, the TRA reported airborne releases of just over 5 million curies. This can be compared with the total reported release of activity from the INEEL of about 12 million curies during this same time period. The MTR played an important role during the early years at the site because of its design, which included aluminum-uranium alloy, highly-enriched, plate-type fuel elements and light water as moderator and coolant. The MTR's normal operations and experiences with cladding failures and other incidents are fairly well documented (De Boisblanc 1958).

Beginning in the early 1970s, the flow rate and radioactivity levels for various radionuclides were monitored and reported monthly. In general, particulate activity accounted for less than 5% of the activity released. At that time, the effluent passed through a continuous tape filter to determine gross particulate and alpha particulate activities, a gamma detector for the total gaseous activity, and a charcoal filter for iodine determination. In the early 1970s the detection limits of the effluent monitors ranged from 6×10^{-8} to 1×10^{-12} microcuries per cubic centimeter ($\mu Ci\ cm^{-3}$).

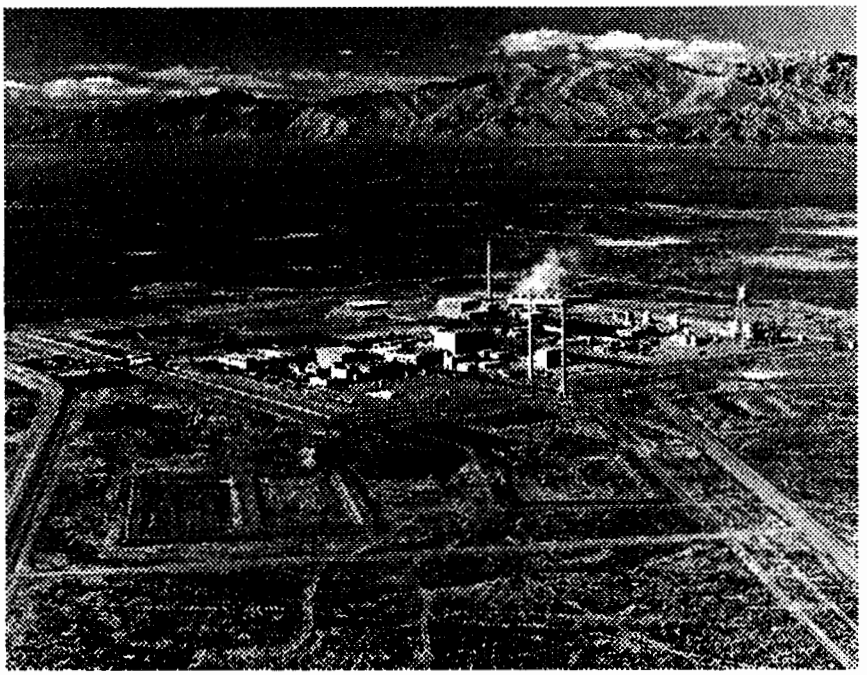

Figure 5. Test Reactor Area at the INEEL where the Materials Testing Reactor, the Engineering Testing Reactor and the Advanced Testing Reactor are located.

The TRA contributed the highest levels of radioactivity in liquid wastes (although the ICPP discharged the greatest volumes of liquid wastes). The major contributors to the total activity released to the TRA ponds were ^{51}Cr and ^{3}H. Most liquid effluents from the TRA come from water purged from the two main reactor primary water systems and from canals. The liquid effluents consisted of

- cold wastes
- chemical wastes from the demineralizers and water softeners, and sanitary wastes
- warm wastes, which contained a small amount of radioactivity but did not exceed the discharge limits for the time
- hot wastes, which were too radioactive for immediate disposal to the groundwater.

Idaho Chemical Processing Plant

The ICPP (Figure 6) recovered enriched uranium and plutonium from spent fuel from reactors at the INEEL, from the U.S. Navy's ship propulsion reactors, and from other research reactors, including some in foreign countries. In the early 1970s, fuels from about 40 reactors were stored or waiting to be processed at the ICPP (ERDA 1977). The ICPP had established processes to handle uranium, aluminum, zirconium, or stainless steel clad elements. The fuel elements were dissolved in an appropriate solvent and the fission products and alloying metals were separated from the uranium in several stages of solvent extraction (Ayers and Burns 1960).

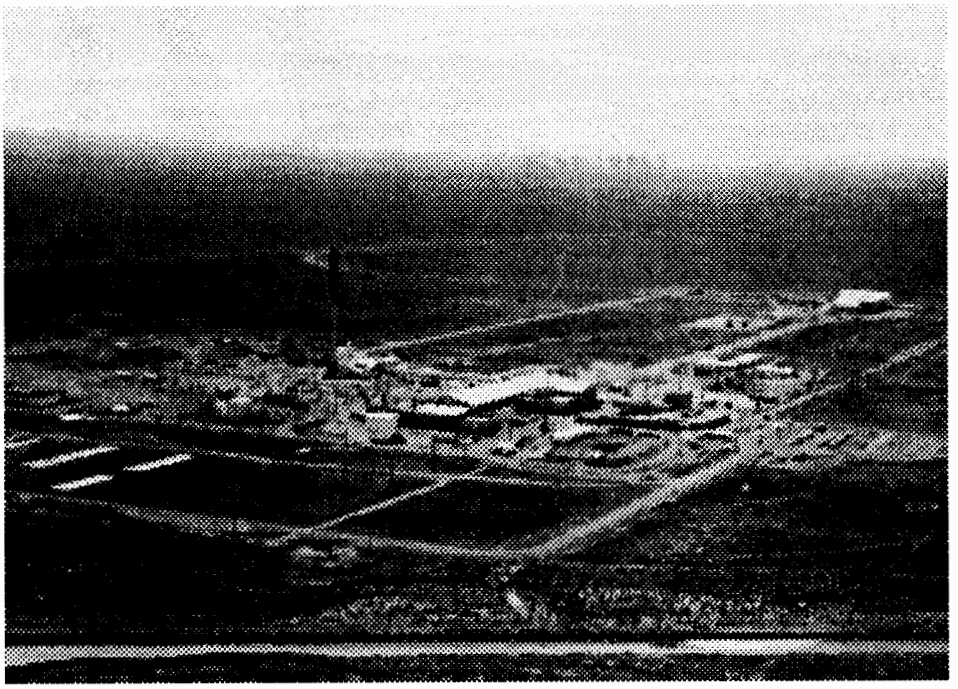

Figure 6. Idaho Chemical Processing Plant at the INEEL. Fuel was received at the building in the upper far right and transported to the main process and laboratory building, the long white building in the center. The 200-foot stack can be seen just to the left of center.

The process began with dissolution in acid, producing uranyl nitrate and nitrates of various fission products and some transuranics. This step was followed by solvent extraction to separate uranium from the fission products, with the final product being uranyl nitrate free of impurities and fission products. The important areas within the ICPP include the Fluorinel Dissolution Process and Fuel Storage Facility, where radioactive spent fuel was stored underwater and where fuel was dissolved and reprocessed; the Waste Calcining Facility (WCF), which converted high level liquid waste into granular solids that were stored in stainless steel-encased concrete binds, the ICPP processing corridors for fuel reprocessing; and the High Level Waste Tank Farm with 11 underground liquid storage tanks.

Airborne releases of radioactivity from the ICPP came primarily from the process dissolvers, waste solvent burner, analytical facilities, and the WCF. Calcination conditions determined the size and concentration of solids in the effluent. Various off-gas studies at the ICPP provided on-going data on the operations of the effluent treatment systems, such as the venturi scrubbers and electrostatic precipitatiors (Wheeler 1959; Cederberg and Bower 1959). There were three off-gas streams, each a treatment system consisting of a reflux condenser and entrainment separator, a demister, and a high-efficiency particulate air (HEPA) filter. All off-gas streams were discharged through the main 250-foot stack to the environment. The liquid waste storage tanks were also vented to this stream. Airborne effluents from the WCF occurred primarily through the process off-gas through the ICPP stack as well. While some airborne releases occurred from the solids storage bins and through sample handling operations, these sources contributed negligible amounts (Lakey et al. 1963).

The WCF was primarily a liquid waste treatment facility. There is extensive and detailed information available on the ICPP airborne waste treatment systems if future detailed dose reconstruction work is necessary (Wheeler 1959; Cederberg and Bower 1959; Ayers and Burn 1960; Wong and Roberts 1979). The reported airborne releases from the ICPP during this 20-year period was 7 million Ci with 6 million Ci released between 1957 and 1963 (ERDA 1977) (Figure 7).

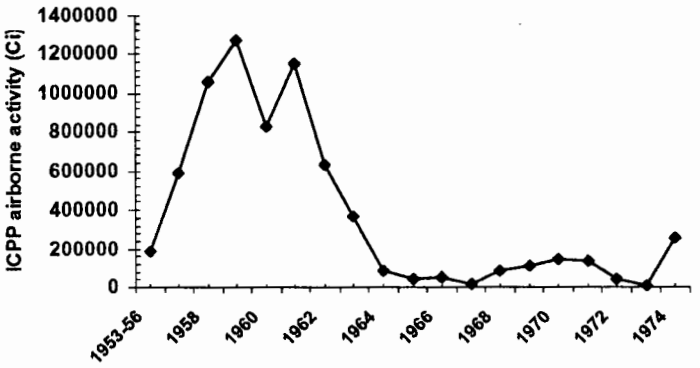

Figure 7. Reported annual releases from the ICPP from 1953–1974, a time period that includes releases from the RaLa processes (ERDA 1977). Annual releases exceeded 500,000 Ci from 1957 through 1962, with peaks of 1.1 million Ci in 1959 and 1.3 million Ci in 1961. The release pattern and magnitude of releases from the ICPP show that the ICPP contributed the majority of activity to the total site releases (see Figure 6).

While actual measurements of all radionuclides were not made on the airborne effluent, the release rates of various fission products were calculated from operational data. From February 1957, measurements of ^{131}I and "beta emitters minus iodine" were calculated based on chemical analyses of stack-gas-monitor scrubber solution samples (Hayden 1957-1963). Beginning in May 1958, daily releases of both ^{132}I and ^{131}I were reported. In the early 1960s, it was reported that over 97% of the activity released from the ICPP was due to tritium and ruthenium and "totals about 15 Ci per day when calcining 200-day cooled waste and about 15 Ci per day when calcining five-year cooled waste" (Lakey et al. 1963).

The ICPP carried out particulate cleanup studies during actual ICPP WCF tests to determine release rates of other important fission products to airborne releases like ^{147}Pr, ^{137}Cs, ^{90}Sr, and ^{144}Ce. The release rate of ruthenium was estimated from pilot plant data and was reported to be less accurate. The early release estimates were calculated for calcining wastes containing fission products aged 200 days or 5 years. For wastes aged 5 years, approximately "8×10^{-5} pounds of particulate solids" were released per hour of operation, contributing about 0.06 curies per day (Ci d^{-1}) of fission products other than ruthenium and tritium. For wastes aged 200 days, the calculated activity release rate increased to 1.6 Ci d^{-1}. The ruthenium release rate was not assessed during these early studies because it was partially volatilized in the calciner (Lakey et al. 1963). However, a decrease in ruthenium activity of "at least 3000 (3 across the calciner and the scrubbing system and 1000 across the silica gel beds and filters)" was assumed in the WCF. This assumption resulted in an estimated ruthenium release rate of 0.12 Ci d^{-1} when calcining wastes containing fission products aged five years (primarily ^{106}Ru) and 34.6 Ci per day for fission products aged 200 days (primarily ^{103}Ru). A 55 Ci ruthenium release from the WCF on October 16, 1964 consisted of >93% ^{106}Ru (see Episodic Releases section). For the reported annual release estimates, the ICPP determined the total activities for ^{90}Sr, ^{106}Ru, ^{137}Cs, ^{144}Ce, ^{147}Pm, and ^{239Pu}, ^{237}Np, ^{60}Co, ^{95}Zr, ^{89}Sr, ^{91}Y, ^{95}Nb, ^{103}Ru, and ^{141}Ce based on these calculated release rates and the calcining history for the year. For ^{3}H, it was assumed that all tritium in the feed was released to the stack with the off-gas.

Before 1975, all ventilation air from the process area was discharged to the stack without treatment. In 1975, an Atmospheric Protection System (APS) was installed to provide continuous filtration of all building ventilation air from process areas and backup filtration of all process off-gases before release to the atmosphere. The system consisted of a 7-foot deep fiberglass prefilter in series with separatorless HEPA filters. High leel radioactive liquid wastes at the ICPP were collected in the tank farm before solidification in the WCF. All other radioactive liquid wastes were collected, processed, and discharged to the disposal well. A continuous liquid waste monitor was installed in 1954 in the ICPP (King 1956).

Liquid waste streams were generated from all areas of the ICPP and varied in volume and degree of contamination. The high level waste streams were sent to the WCF. Two intermediate and low-level waste streams were collected in the evaporator tank before discharge to the injection well. It was reported that the cell floor-drain collection system for low-level waste handled about 200,000 gallons per year and the process equipment waste collection system for intermediate level waste handled about 1 million gallons per year. Additional liquid wastes totaling about 15,000 gallons per year from other areas at the INEEL were handled in this system as well (Dickey et al. 1972). The most abundant radionuclides measured in liquid waste discharged to the disposal well were tritium, ^{137}Cs, and ^{106}Ru, and ^{90}Sr.

Test Area North

Test Area North (TAN) (Figure 8) was built to support the Aircraft Nuclear Propulsion (ANP) Program and was activated in 1954 for the U.S. Air Force for testing nuclear engine prototypes and investigating the feasibility of a nuclear or chemical propulsion system for military use. This program conducted numerous Initial Engine Tests (IETs) during the late 1950s until 1961 when the program was cancelled. The Technical Services Facility (TSF), at the center of TAN, provided support for the area. The only reactor operations at the TAN complex occurred at the Loss of Fluid Tests (LOFTs) facility. These programs are evaluated and discussed in the Episodic Release section of this report.

Figure 8. .Aerial view of part of the Test Area North, location of the Aircraft Nuclear Propulsion Program and the Loss of Fluid Tests. Test Area North was a minor contributor to overall site operational releases.

The Low Power Test (LPT) and the Experimental Beryllium Oxide Reactor (EBOR) facilities complex are located about 2 miles from the main TAN support facilities. These facilities were originally constructed for reactor testing activities during the Aircraft Nuclear Propulsion program. The two facilities shared a deep well, pump and two water storage tanks with a combined capacity of 195,000 gallons, but the water system was not used heavily (ERDA 1977).

Releases of airborne effluents from eight areas within the TAN support facilities pass through filtration systems before discharge. Operational releases from the TAN totaled approximately 54,000 Ci from 1958-1974, most attributed to the ANP Program and IETs, which are treated as episodic releases in this report. This value (54,000 Ci) can be compared to total reported releases from the INEEL of 10 million Ci during this same time. It should be stressed that comparisons of total activity released can assist with determining the relative importance of

activities at different facilities; however, it is important to also assess the quality of the release and evaluate individual radionuclide contributions because of the large differences in dose contributions from different radionuclides. The TAN area is an important contributor to episodic releases but is of less importance as a routine airborne release source.

A radioactive liquid waste system collected and processed intermediate-level radioactive liquid wastes generated in the TAN area and transferred it to one of three underground 10,000-gallon stainless steel collections tanks (Kerr 1971). The liquid wastes were concentrated in an evaporator and the concentrated solution pumped to one of two 50,000-gallon underground liquid waste feed tanks. Solids were separated and sent to the Radioactive Waste Management Complex (RWMC) at the INEEL. During 1969 the waste collection and evaporation rate at TAN averaged about 14,000 gallons per month (Kerr 1971). The LOFT tests periodically produced large volumes of liquid wastes, but these were sent to the ICPP for processing. Originally, the liquid effluent was combined with low-level radioactive liquid waste and discharged to a disposal well. In 1972, the site replaced the disposal well with a disposal pond, an unlined diked area encompassing about 35 acres that could receive about 33 million gallons per year (ERDA 1977). From 1959–1974, TAN reported liquid effluent releases to the disposal well or pond of 58 Ci with highest releases in 1959, 1968 and 1969. This activity can be compared to the total activity reported in liquid wastes from 1959–1974 of over 50,000 Ci (see LiquidEffluents.xls).

Argonne National Laboratory-West

Argonne National Laboratory-West (ANL-W) (Figure 9) was established in the mid1950s to operate three major reactors: Transient Reactor Test Facility (TREAT) in 1959, the Experimental Breeder Reactor No. 2 (EBR-II) in 1961, and the Zero Power Plutonium Reactor (ZPPR) in 1969. It also houses the Hot Fuel Examination Facility (HFEF) and the laboratory and support complex. EBR-II is an experimental liquid-metal cooled fast breeder reactor that was unmoderated and submerged in the primary tank filled with about 90,000 gallons molten sodium (ERDA 1977).

The major release point to air is a 200-foot high stack, centrally located in the ANL-W area that received airborne effluent from the EBR-II building, the HFEF, and the service buildings. Radioactive airborne effluent from the EBR-II complex passed through HEPA filters, through a radiation monitor, to the 200-foot exhaust gas stack. The flow through the stack averaged 70,000 cubic feet per minute and all discharged air from the stack was monitored (Hogg et al. 1971b). The principal radionuclides identified in the stack effluent were tritum, ^{41}Ar, ^{85}Kr, and ^{133}Xe. All airborne effluents from the fuel assembly and storage building was treated as radioactive.

The TREAT reactor was designed "to produce short extreme pulses of nuclear energy with resultant temperature high enough to permit meltdown studies of selected prototype and experimental fuel elements." All gaseous effluent from the reactor was exhausted through a bank of six parallel HEPA filters and discharged into a 60-foot high exhaust stack. ZPPR was designed to provide information for designing and developing large plutonium fueled fast breeder reactors. It allowed fuel loading in a variety of patterns to simulate various reactor core designs. Airborne effluents were reported to be monitored for beta-gamma-emitting particulates by forced flow, fixed filter continuous air monitors both upstream and downstream of the HEPA filters. The downstream flow was monitored continuously by an alpha monitor as well (ERDA 1977).

Reported annual airborne releases from the ANL-W are located in FacilityAirReleases.xls. Airborne releases of radioactive fission gases were highest from 1965 through 1969 because the

Fuel Cycle Facility that processed EBR-II fuel lacked an adequate holdup system to reduce the release of short-lived radionuclides like ^{133}Xe and ^{135}Xe. In 1969, the Fuel Cycle Facility was shut down (Hogg et al. 1971b).

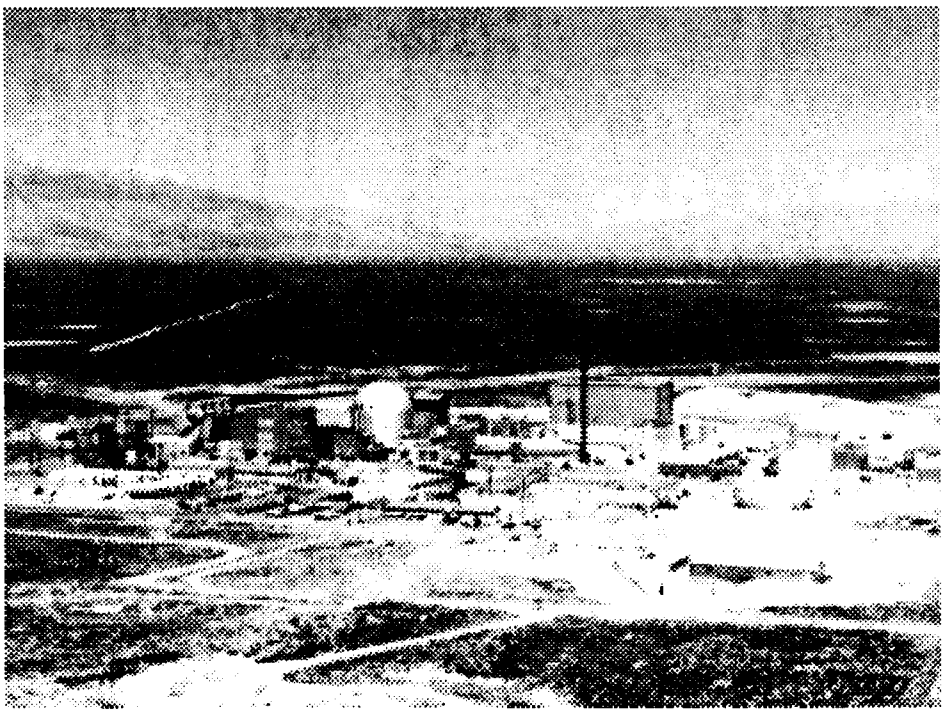

Figure 9. Aerial view of Argonne National Laboratory-West. The 200-foot airborne effluent discharge stack is visible to the right of center.

Liquid radioactive wastes form the ANL-W came primarily from the EBR-II Area. Liquid wastes identified as low activity were piped to a 35,000-gallon underground seepage pit (37 feet long × 18 feet wide × 10 feet deep), located outside the fenced area of the EBR-II site, until 1973 when the pit was covered with 8 inches of concrete (Hogg et al. 1971b). Low level radioactive liquid waste was released to a seepage pit Most of the low level liquid wastes originated in the Fuel Cycle Facility or in the laboratory building. Osloond (1970) reported that over 76,000 gallons of low level liquid waste, containing mainly $^{141,144}Ce$, ^{137}Cs, $^{58,60}Co$, $^{106}RuRh$, ^{54}Mn, ^{51}Cr and $^{95}ZrNb$, were discharged to this underground pit in 1970.

The intermediate activity liquid wastes were transported through underground pipes or by portable tanks to retention tanks at a waste evaporator at the laboratory building and discharged into one of two 2900-gallon carbon steel settling tanks. The concentrated bottom material from the evaporator was encased in concrete inside a steel drum and buried at the RWMC. The highly radioactive wastes, generated in the shielded cave areas of the ANL-W analytical laboratory, were sorbed on vermiculite in quart jars and also buried at the ERB-II underground storage facility (Hogg et al. 1971b).

The Industrial Waste Pond, an unlined seepage pond excavated in 1959 to a depth of 4 m (13 feet) and an area of 3 acres, was designed to receive non-radioactive wastes, but the effluent was

monitored continuously (ANL-W 1973). Activity released in liquid wastes at ANL-W totaled abut 3300 Ci from 1961-1974, compared to over 50,000 Ci released in liquid wastes from all areas onsite for that same period. Radioactive solid waste from the ANL-W facilities was stored at the ANL-W Radioactive Scrap and Waste Facility or sent to the RWMC.

Naval Reactors Facility

The Naval Reactors Facility (NRF) operated three naval reactor prototypes:

- the S1W, the prototype for the nuclear powered submarine Nautilus, beginning in 1953,
- the A1W reactor, the prototype for the aircraft carrier Enterprise, beginning in 1959, and
- the S5G reactor, used to train Navy personnel after beginning operations in 1966; water flow through the reactor was by thermal circulation rather than by pumps.

The Expended Core Facility (ECF) opened at the NRF in 1958 to examine, and test components that had been irradiated in nuclear reactors, and also structural materials removed from expended naval core fuel modules. (ERDA 1977). After separation the structural parts were transported to the RWMCx, and the expended fuel was sent to the ICPP for reprocessing.

Airborne radioactivity at the NRF occurred primarily when reactor coolant systems were drained or sampled. All air passed through HEPA filters or charcoal filters, with continuous monitoring at the ECF where the expended naval reactor core components were handled (ERDA 1977).

Liquid radioactive wastes were processed separately in the four faculties at the NRF, and the effluent was discharged to two seepage basins that were fenced pits in the earth covered with rock or dirt. From 1953–1974, NRF reported liquid effluent releases to the seepage basins of just over 400 million gallons. The reported total activity during this 20-year period was 340 Ci, excluding tritium, and about 9 Ci of tritium (ERDA 1977). In addition to tritium, the major radionuclides in liquid waste released to seepage basins from the NRF were ^{14}C and ^{60}Co.Releases were highest from 1958 through 1964. Hogg et al. (1971b) also reports annual releases of liquid wastes and quantities of selected radionuclides to the NRF Seepage Pond during the 1960s. Nonradioactive industrial wastes from the NRF were discharged through culverts to a dredged drainage ditch located northwest of the NRF. The sanitary wastes were discharged to 2 seepage lagoons with a surface area of over 8 acres. In the 1960s, approximately 1.8 million gallons per month were discharged to the lagoons.

Other Facilities

The Central Facilities Area (CFA) provided numerous support services for other operational areas at the site. Up until the late 1970s, the area of interest as a potential radioactive release source was the laundry for radioactively contaminated clothing that generated about 340,000 gallons per month of liquid waste. Off-gas from the dryers was screened and vented to the outside but this was not monitored (ERDA 1977).

The Experimental Breeder Reactor No. 1 (EBR-I), the first nuclear reactor at the INEEL site located near the CFA, achieved criticality in 1951 and operated until 1964 (DOE 1991a). It was

unmoderated and used sodium-potassium as coolant and enriched uranium as fuel. The EBR-I core meltdown was evaluated as an episodic release. The 5 BORAX reactors at the INEEL established the boiling water reactor technology where the coolant moderator boils in the reactor core and passes saturated steam directly to the turbine for power generation. The BORAX-I test reactor operated from 1953-1954 and the BORAX-I excusion was treated as an episodic event. The other BORAX reactors were the BORAX II, which began operation in 1954 at a power level of 6 MW(t), the BORAX-III reactor operated in 1955, the BORAX-IV operated from 1956-1958, and, finally, the BORAX-V reactor operated from 1962-1964 at a power level of 40 MW(t). The testing of the BORAX-IV Reactor between March 11 and 27, 1958 was evaluated as an episodic release event.

The Power Burst Facility (PBF) was a high performance, water-cooled uranium oxide fueled reactor designed to provide information on light-water reactors. Airborne effluents were filtered and passed through charcoal beds to remove iodine. Liquid wastes were pumped to a disposal well or held in tanks for transport to the ICPP.

The Auxiliary Reactor Area (ARA), east of the Central Facilities Area (see Figure 1), included several areas where U.S. Army portable power reactors were tested until about 1965. ARA-I and ARA-II were originally the location of the Stationary Low Power Reactor (SL-1) until the SL-1 accident in 1961 (see SL-1 Accident in Episodic Releases section).

Numerous special government programs were conducted over the years at the INEEL. One such program was the ANP Program (1953-1961), which included the IETs. These tests were conducted at TAN for the U.S. Air Force to assess various nuclear engine prototypes. Other programs at the INEEL were the Fission Product Field Release Tests (FPFRT) conducted for the U.S. Air Force to assess radioactivity from potential accidents involving nuclear powered aircraft from July to December 1958; the Special Power Excursion Reactor Test (SPERT); and Controlled Environmental Release (Radioiodine) Test (CERT) that included over 30 intentional, planned releases of radionuclides to study environmental transport and uptake from May 27, 1963 through December 1977. It was important to assess the Boiling Water Reactor Experiment (BORAX) Program and the RaLa Program for their contribution to radionuclide releases. All unplanned or episodic non-routine events are evaluated in the Episodic Release section.

In the next section, we discuss the RaLa program conducted at the ICPP using fuel elements from the Material Testing Reactor from 1956 through 1963. Releases from the entire program were evaluated as part of the routine releases. In addition, individual "runs" within this program that resulted in high, short-term releases were evaluated as episodic events.

Radioactive Lanthanum (RaLa) Program

Background

The RaLa facility at INEEL was designed and built to replace an outmoded ^{140}Ba production facility at Oak Ridge National Laboratory in Tennessee, which had operated from 1945-1956. During this time (late 1940s-early 1950s), there was increasing demand for higher specific activity barium, which decays to radioactive ^{140}La, an intense radiation source. This demand resulted in the search for a location with a source of higher specific activity ^{140}Ba. This search led to the INEEL, because of the availability of high specific activity fuel from the fairly new MTR. Oak Ridge National Laboratory had achieved a maximum production of about 10,000-Ci batches

and had attempted production of up to 30,000-Ci batches. In October 1952, the decision was made to proceed with a long-range RaLa production program at ICPP, with essentially no limits on the amounts of material desired (30,000- 100,000 Ci batches) (Legler et al. 1955). At the same time Hanford Works in Washington was involved in designing iodine scrubbers and dealing with problems that arose in the RaLa process.

The program at the INEEL included three phases:

1. Pilot Plant;
2. Testing with unirradiated MTR fuel;
3. Active testing with irradiated fuel.

From November 1, 1955 through June 30, 1957, a startup program, located in Cell L of the ICPP, was conducted from plant takeover to successful high activity level production. During the document search, we located the original handwritten logbooks for the RaLa Program with entries from 11/29/53 through 1/5/68, day by day, month by month (Figure 10). These logbooks listed ^{131}I releases from each RaLa Program "run", with several pages of calculations for each run, along with stack flow rates. In some months there are 3 to 4 orders of magnitude variation in ^{131}I released to atmosphere. The logbooks also recorded ICPP stack releases and liquid effluent releases to injection wells. The results of pilot plant testing of the RaLa process, laboratory research and development on the process for separation of ^{140}Ba from MTR fuel elements, and regular exchanges about the process are well documented (Andersen and Weech 1954; Andersen et al. 1959). These early reports deal with chemical flow sheet development, data on the feasibility of barium-strontium separation using the chromate method in the presence of high radiation fields, and the results of laboratory and pilot plant corrosion investigations in the Multicurie cell at the ICPP.

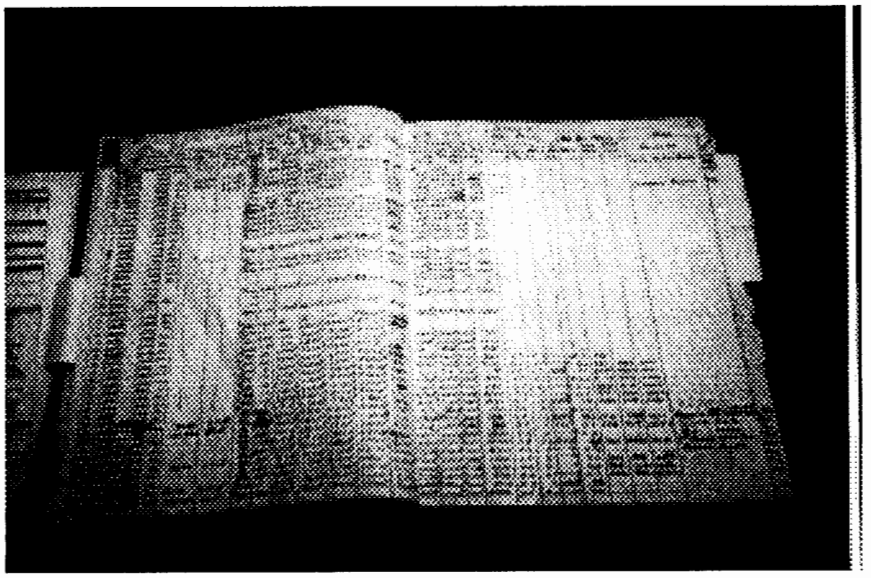

Figure 10. Photograph of a handwritten logbook like those that recorded information of operations during the RaLa program.

Because ^{140}Ba has a fairly short half-life of 12.8 days, the fuel was processed as soon as possible (usually about 36 hours) after removal from the MTR to maximize the yield of the decay product, ^{140}La. In contrast, during normal operations, fuel reprocessing at the ICPP was performed only after fuel had been out of the reactor for 120 days, allowing for significant decay of gaseous fission products with short half-lives. For the RaLa program, however, fuel was reprocessed as soon as possible after removal from the reactor (normally 2-7 days).

These runs resulted in large releases of fission products to air, with ^{131}I (half-life of ~8 days) of most concern. There were about 78 separate runs from 1957 through 1963, with releases occurring over a couple of days to weeks. Another fission product ^{132}Te (half-life of ~78 hours) decayed to ^{132}I (half-life of 2.5 hours), which contributed much of the activity several days after processing and subsequent release of the iodine when solutions are jetted or released. The ratio, ^{132}I: ^{131}I, from the RaLa process releases was ~ 3.3.

All RaLa runs were well documented and both effluent sampling at the stack and some environmental monitoring were done. Discharges of effluent to the stack were measured by bubbling small fractions of stack effluent through a liquid scrubber. The "beta minus iodine activity" was obtained by evaporating a portion of the liquid sample and gross counting the remainder for beta activity. Quarterly technical progress reports from the ICPP summarized the details of the process operations and activities for that time period (e.g., Stevenson 1957). These reports provide insight into the development of methods for modifying the process for efficiency and for reducing releases of iodine during the RaLa process. Based on the known chemistry of iodine compounds, it was initially thought that the radioactive iodine released from an MTR element during the RaLa process would remain behind, either combining with the sodium hydroxide in the caustic scrubber solution to form sodium iodide, or collected in a 10,000 ft^3 gas holder and held for release "under favorable weather conditions"(Stevenson 1957). Neither expectation was seen and the operating schedule for RaLa was affected by finding ways to limit the iodine released to the atmosphere. These reports contain results and details about the distribution of ^{131}I in the RaLa process streams, which may need careful review if additional investigation is required for the RaLa runs.

Uncontrolled releases of large amounts of iodine beyond the design specifications occurred because the original iodine containment was in place only for the process off-gas and not for two other parts in the RaLa process system: the vessel off-gas and cell ventilation air (Cederberg and MacQueen 1961). Concern led to continued efforts to reduce iodine releases. In 1961, the scrubber solution was changed from sodium hydroxide to nitric acid with mercury salts. The addition of the mercury salts to several process solutions resulted in a 10-fold reduction in iodine concentration in off-gas. Another 10-fold reduction was obtained by installing an activated charcoal adsorption unit in series with the original iodine removal scrubber. However, the factor limiting the overall iodine removal efficiency at this time was the ability to remove iodine-bearing dusts and particulates.

During the INEEL HES quarterly meetings, questions arose regarding the involvement of Hanford in the RaLa process operations at the INEEL. We searched for and reviewed documents from Hanford and the INEEL for information to clarify Hanford's role. The historic record clearly shows that Hanford shipped fuel slugs to the INEEL regularly. On January 14, 1952, a Hanford memorandum indicated that the INEEL had asked if Hanford would be able "to can five hundred ten simulated J slugs for cold runs during the start-up of the Idaho Chemical Processing

Plant." The memo continued that " ...the GE Company (Hanford) believes that they will be able to undertake this service without too much difficulty ..." (HAN 42692. January 14, 1952. Idaho CPP Start-Up Materials from the ICPP). Shipments of 60-day cooled slugs were to begin in March 1954 according to a memo written December 1, 1953 from F.K. Pittman at Hanford to the ICPP: "...60-day cooled slugs shipments will begin in March 1954 and will continue for 6 months at monthly rate of about 70 kg of U-235 in spent fuel elements. After this, shipments will decrease to about 7 ½ kg per month....". these records indicate that fuel was shipped to the INEEL, but after it had been cooled for weeks or months. Fuel for the Ra La runs was cooled at the most for about 2 days to limit the decay of the ^{140}Ba. These record reviews indicate that Hanford did not supply fuel elements for the RaLa processes at the INEEL.

Table 1 is an example of the data found in a series of reports detailing the number and types of fuel shipments to the INEEL.

Table 1. Slug Shipments from Hanford to the ICPP

	Number of slugs shipped		Number of casks shipped	
Shipment period	C	J	C	J
March, April 1956[a]	3360	3360	24	24
November 1956[a]	4760	280	34	1
May, June 1957[a]	5320	224	38	2
May, June 1958[b]	5600		40	
October, November 1958 (final)[b]	2700	224	20	2

[a] From HAN-61940 (P.G. Holstad, P.G. March 26, 1956).

[b] From HAN-68946 (R.E. Johnson, May 6, 1958); shipments left Hanford on Monday and Tuesday.

Documentation of RaLa Releases

We compiled the daily reported releases of ^{131}I (and ^{132}I when it was reported) from March 11, 1957 through June 14, 1963 to determine the best approach to screening the releases from the ICPP during the RaLa runs. We carefully reviewed previous analyses of RaLa runs (DOE 1991a) to determine whether the RaLa operations should be treated as part of the chronic operational releases or as episodic releases. Other reports showed that more ^{131}I was being released for days following the RaLa runs than was being released during the several hours of the RaLa runs themselves (Legler et al. 1957). This point is illustrated in Figure 11 and is discussed below.

Daily releases of ^{131}I and beta-minus iodine activity were reported beginning on July 7, 1957. When the weekly ^{131}I releases in 1957 and 1958 from the ICPP are plotted, the RaLa runs responsible for the largest releases are easily seen (Figure 11). RaLa run #2 released the largest quantity of ^{131}I (220 Ci) reported for that week. RaLa run #1 released a reported 145 Ci ^{131}I for the week. Interestingly a high release of ^{131}I (about 50 Ci) on Saturday, March 1, 1958 was not associated with a RaLa run and points out the difficulty of completely separating out the releases from the RaLa operations from other processing activities at the ICPP during that time. The carbon (charcoal) beds were installed and used on August 6, 1958 to remove iodine from the airborne effluent, and this marks a point when iodine releases decreased considerably. The effluent monitoring results and daily and monthly release estimates reflect this impact (see FacilityAirReleases.xls).

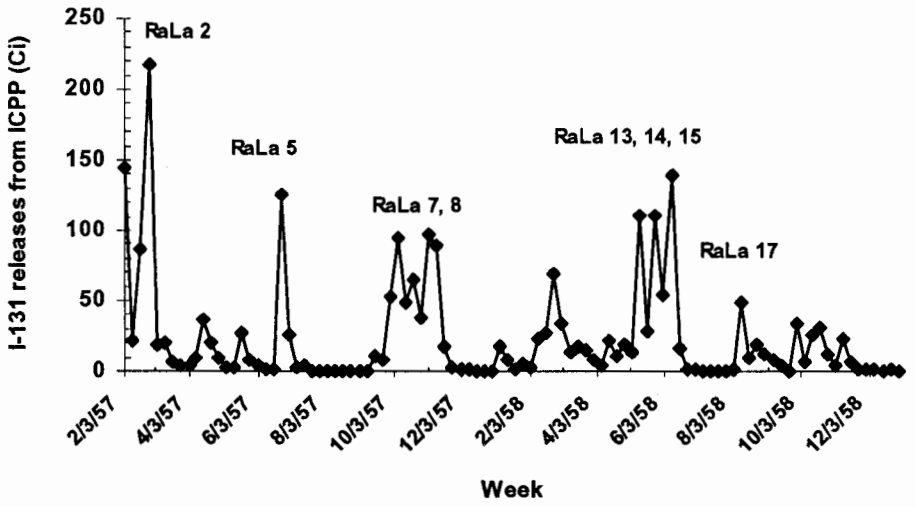

Figure 11 Reported weekly releases of ^{131}I from the ICPP (Hayden 1957-1963). Some of the RaLa runs with the highest reported releases are identified. RaLa run #2 released the largest quantity of ^{131}I (over 200 Ci) reported for that week. RaLa run #5 released a reported 145 Ci ^{131}I for the week.

Figure 12 shows the daily releases from October 1, 1957 through mid-February 1958 and illustrates that ^{131}I releases could occur for up to several days after a RaLa run was complete. For RaLa run 7, the highest releases occurred on the day of the run (~41 Ci). RaLa run #8 occurred on October 21, 1957 but the highest release following that run was on October 23 (~26,000 mCi). Approximately 3500 mCi was released on the day of RaLa run #9, but over 15,000 mCi was released over the next 5 days. A similar release pattern was seen after RaLa run #10. These daily release records support our approach of including the releases as chronic operational ICPP releases.

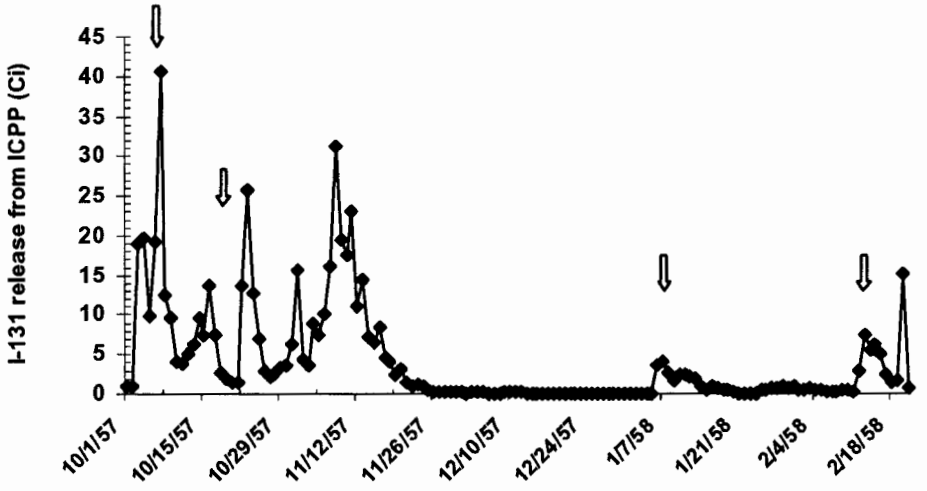

Figure 12. Daily releases of ^{131}I from the ICPP from October 1, 1957 through February 15, 1958. The block arrows indicate the occurrence of RaLa runs 7 (October 7, 1957), 8 (October 21, 1957), 9 (January 6, 1958), and 10 (February 15, 1958). For RaLa run 7, the highest releases occurred on the day of the run (~41,000 mCi). However, the highest releases following RaLa run 8 occurred 2 days later on October 23 (~26,000 mCi).

While the daily release record for ^{131}I from the RaLa runs in the weekly and daily reports (Hayden 1957-1963) does support those summarized in DOE (1991a), there are a few minor discrepancies. In DOE (1991a), the release for October 21, 1957 (234 Ci) actually reflects releases for that entire month. This observation is supported when the daily release values from the daily release records are summed to obtain a monthly total of 278 Ci (Hayden 1957-1963). DOE (1991a) reports that RaLa run #15 occurred on June 2, 1958, while the daily discharge report stated that RaLa run 15 occurred on May 28, 1958 and no RaLa run was noted for June 2.

For the screening calculations, we have included all RaLa runs in the annual release totals from the ICPP. However, several of the RaLa runs conducted before the charcoal beds were installed released significant quantities of iodine to the atmosphere. These RaLa runs were also evaluated as episodic events (see Episodic Releases section).

SCREENING ROUTINE RELEASES

To determine the INEEL facilities that may have impacted persons offsite to the greatest extent, we reviewed and evaluated a wide range of available documents and reports to obtain source term estimates for use in the screening method. These documents included the Idaho National Engineering Laboratory Historical Dose Evaluation (DOE 1991a); annual ICPP and site monitoring reports (e.g., Honkus 1982); quarterly ICPP effluent monitoring reports (e.g., ENICO 1983a, 1983b); a series of weekly ICPP notegrams reported daily releases to air of ^{131}I, ^{132}I and the approximate beta particulate emitters other than iodine from the main ICPP stack gas monitor stack (Hayden 1957-1963); data from the Radioactive Waste Management Information System (RWMIS) (e.g., Litteer et al. 1991); annual site environmental monitoring reports (e.g. INEL 1979; Hoff et al. 1984); and numerous site memoranda and reports (e.g., Hayden and Rich 1958-

1959). Releases from the INEEL facilities are discussed separately for releases to air, discharges of liquid effluents, and disposal of solid radioactive wastes.

Confirmation of Reported Radionuclide Releases

We compiled information on annual releases of radioactive materials in airborne and liquid effluents and data on solids buried at the RWMC for certain periods of operation. We also compiled monthly, weekly, and daily releases of certain radionuclides for certain facilities that released significant levels of radioactivity. We used the monthly, weekly, and daily data for selected time periods for comparisons to annual release estimates to confirm that the reported annual releases accurately reflected the original release data. This exercise provided some confidence that the annual release data obtained from annual environmental and effluent release reports, from the RWMIS data, and from the Historic Dose Evaluation (DOE 1991a) documents would provide a good foundation for our screening calculations. For example, Table 2 compares the annual reported release of ^{131}I and ^{132}I from data compiled from various types of reports.

Table 2. Comparison of Daily, Monthly and Annual Reported Releases (Ci) of ^{131}I and ^{132}I from the ICPP

Radionuclide	1957[a]	1958	1959	1960	1961
^{131}I					
Daily[a]	1347	1025	239	32	
Monthly[a]	1347	1630	223	28	23
Annual[b]	1400	1000	224	32	42
^{132}I					
Daily	nr	2012[c]	2074[d]	172	
Monthly	nr	2628[c]	2074[d]	201	226
Annual	4000	3380	1550	176	227

[a] From Hayden 1957-1963.

[b] From DOE 1991a.

[c] From May 25, 1958 onward

[d] Does not include the reported release of 9780 Ci ^{132}I on October 16, 1959 from a criticality event; see Episodic Release section.

In checking for consistency between monthly, quarterly, and annually reported releases for various radionuclides, agreement was generally good. A few cases of incorrect math were noted, such as in the total reported release estimate for ^{238}Pu to air. In the final tally of releases of ^{238}Pu to air through the main stack at ICPP, the annual total was an order of magnitude low. In combining the monthly release estimates, the annual total was 1.45 mullicuries (mCi), not 0.145 mCi as reported (ENICO 1983b). For releases of ^{137}Cs in liquid effluents to the ICPP injection well, the annual total was reported as 61.2 mCi in the quarterly effluent report. It should be 63.2 mCi (WINCO 1984). This error was caused by transposing the value for the 1st quarter total; the reported total given as 21.6, when it should have been 26.1 mCi., which is the total of monthly values. For ^{89}Sr, the annual total was reported as 5.65 mCi, when it should have been reported as

9.56 mCi based on monthly totals (WINCO Feb 1984). This reporting error occurred in carrying the 1st quarter total from the quarterly effluent report (ENICO 1983a), which was given correctly based on monthly totals for January, February, and March 1983 as 7.19 mCi, to the annual totals for 1983. In the 1st quarter report, the ^{89}Sr total was given correctly as 7.19 mCi; however, in the annual summary report, the total for 1st quarter was given as 3.27 mCi.

Krypton-85 releases, the largest activity releases at ICPP, were normally reported only if the total release during a month exceeded 50 Ci. During early 1983, the fuel processing and the rare gas recovery plant were not operated; thus, there were no reportable releases of ^{85}Kr. Beginning in April 1983, all measurable ^{85}Kr releases were reported in effluent monitoring reports (ENICO 1983b). The high ^{85}Kr releases in April and May were associated with "the venting and recovery of gas from several previously filled cylinders. These cylinders contained impurities and were used in training new operators as well as to recover the krypton-85." The highest releases occurred on April 25 when over 365 Ci was released.

Toward the end of 1984, an ambient air monitoring program for particulate matter was initiated to comply with U.S. Environmental Protection Agency (EPA) regulations. High-volume samplers were placed around the ICPP and the filters were collected and analyzed weekly for total suspended particulates matter and then scanned for gamma activity (WINCO 1985). The first data from the gamma scans showed that the two most common radionuclides found in ambient air were ^{137}Cs and ^{7}Be although it was reported that the levels did not exceed the control guides for uncontrolled areas at that time.

Evaluation of Airborne Operational Releases over Time

We evaluated the releases of radioactivity over time to identify those years during which airborne operational releases were highest. Gaseous and particulate radioactive material discharged from the ICPP main stack were sampled at the 90 feet (27 meters) level and radionuclide-specific analyses were done after 1960. A stack sampling probe was installed in 1957 to monitor radioiodine released from the processing of 2-day cooled fuel elements from the MTR for the RaLa Program. In 1960, the system was improved to provide measurements of primary fission products as particulates. In 1979, it was again modified to update the flow measurement instrumentation and the sampling system for gaseous radionuclides. The sample probe was a 38-millimeter isokinetic probe permanently installed through a 10-centimeter diameter port in the stack wall.

Ayers and Burn (1960) describe the controls in place in the 1950s to treat and monitor the releases of radioactive effluents to air from the ICPP. They noted that "the RaLa process, which normally uses two-day cooled fuel, are much more hazardous and requires special handling." For later years, Wong and Roberts (1979) provides detailed descriptions and diagrams of the stack sampling probe and monitoring system in place at the ICPP. The stack monitors were calibrated against weekly or semiweekly gas and particulate samples analyzed by NaI gamma spectrometry. The APS was installed in 1975. The main stack line was split and one line passed through a large filter to remove particulates. The second line passed through a smaller filter, where the sample was continuously monitored by a sodium iodide scintillation detector for gamma and high-energy, beta-emitting radionuclides (Bowman et al. 1984). They collected both filters daily and analyzed them for radioactivity and every 5 days analyzed them for ^{89}Sr and ^{90}Sr, and gross alpha. Gross beta activity was based on the gamma scan information, and monthly composite samples

were analyzed for ^{238}Pu and $^{239,240}Pu$. After passing through the filtration, the effluent was monitored for ^{3}H, ^{14}C, ^{129}I, ^{125}Sb, and ^{85}Kr.

Reported releases from the ICPP were highest in the late 1950s, primarily from the RaLa Program when spent fuel elements were processed at the ICPP to recover ^{140}Ba. Barium-140 decays to ^{140}La, which produces a desirable high-energy gamma ray. This processing of short-cooled fuel resulted in relatively large releases of radioiodine during and following dissolution of the elements. Between 1957 and 1959, RaLa process effluent comprised the majority of total plant discharges, potential doses from which were dominated by ^{131}I. Releases were reduced beginning in August 1958, following installation of charcoal beds for iodine removal. Stack releases to the environment were reported as ^{131}I, ^{132}I (beginning in April 1958), and gross beta activity minus iodine.

Until the 1980s, only the Main Stack (CPP-708) from the ICPP was monitored continuously. The New Waste Calcining Facility (NWCF) ventilation stack (CPP-659) and the Graphite Storage Facility (GSF) stack were monitored periodically for radioactive releases. For the first time in 1984, the NWCF ventilation stack emissions were reported in the quarterly and annual reports, and there was some effort in more accurately reporting air volume through the NWCF stack (WINCO 1985). The radionuclides monitored and reported from the NWCF stack were ^{137}Cs, ^{238}Pu, $^{239,240}Pu$, and ^{90}Sr. In 1984, stack monitoring began at the Fluorinel Dissolution Process and Fuel Storage Facility (FAST) dtack and at the Coal-Fired Steam-Generating Facility stack. Data were first reported for releases from FAST in the quarterly reports in1984 (WINCO 1985). The main stack and FAST stack had particulate and gaseous monitoring systems by the mid1980s that operated continuously and used proportional isokinetic sampling. Particulate filters were collected and analyzed daily and the gaseous sampler every 2 weeks. Monthly and annual emissions from the Remote Analytical Laboratory (RAL) stack (CPP-684) were first reported in 1987 (Krivanek 1988). The NWCF ventilation off-gas system and the RAL ventilation off-gas were continuously monitored for radioactive particulates with filters collected periodically.

During this time, tritium accounted for "over 98 percent of the total (excluding noble gases) airborne activity released." Tritium releases were high during late 1982 and early 1983 due primarily to evaporator operation to concentrate medium-level liquid wastes before calcining. Releases of tritium, a fission product associated with the spent nuclear fuel processed at the ICPP, generally paralleled the NWCF; therefore, when the NWCF was not operating, tritium releases tended to be lower. After about 1980, ^{129}I was the next most prevalent radionuclide in the airborne emission, and releases were closely related to process operation. There were high releases of ^{129}I in August 1982, with releases decreasing somewhat during early 1983.

In general, releases (especially from ICPP) tended to decrease during the 1980s because of improvements in recovering radionuclides from the effluent; for example, the operation of the Rare Gas Plant to recover ^{85}Kr led to decreases in airborne releases in 1981 (Honkus 1982). During that same time, however, releases of ^{106}Ru increased because of a breached filter at the WCF and ^{131}I releases increased during September and November 1981 because EBR-II fuel and waste was processed. Plutonium emissions closely followed use of the waste solvent burner in the 1980s (ENICO1983a) and rose sharply in April 1983 to their highest levels since 1979. Increases in plutonium, strontium, and cesium emissions were also attributed to maintenance activities in N-Cell of ICPP-601 that disturbed and released residual plutonium to the atmosphere via the main stack (WINCO 1985). A gradual decrease in plutonium emissions in 1985 from the main stack

was attributed to a change in the chemical analytical techniques, similar to the FAST stack analysis (WINCO 1986).

Reported ^{90}Sr concentrations in the airborne emissions were high in the early 1980s and declined in later years. Part of this difference was due to a change in reporting the strontium data. Before 1983, the measured ^{90}Sr activity was doubled to account for ingrowth of the ^{90}Y daughter. The resulting value was then compared to the Radiation Control Guide for ^{90}Sr. However, this practice was discontinued because the ^{90}Sr Radiation Control Guide already took into account the ingrowth of the ^{90}Y daughter. Therefore, it was not necessary to double the concentrations. This same practice applied to reported ^{106}Ru values; ^{106}Ru activity was doubled to consider ^{106}Rh.

The end result of the release pattern is that even though total releases were tending to decrease year by year, there were small accidents and special processing events in later years that increased releases of certain radionuclides.

Evaluation of Liquid Effluent Releases

The INEEL site has no surface streams or rivers flowing from onsite to offsite locations. The Big and Little Lost Rivers and Birch Creek originate in the mountains to the northwest, and flow to the floodplain to a several hundred acre area called Lost River Sinks onsite (Figure 1). In this area, water recharges the Snake River Plain Aquifer, which lies beneath the site. Flow in the Big Lost River is highly variable, with peak flows occurring in June and July from snowmelt, and generally no flow during winter months. Data show that there was a general decline in flow through the late 1960s and early 1970s, reaching a minimum during 1976-1980, with no flow from 1977 to mid 1980 (Hull 1989). Several nuclear reactor facilities and the RWMC are located on the floodplain of the Big Lost River and an onsite diversion dam on that river was built to regulate its flow after several floods inundated the site and caused problems at the RWMC. The diversion dam also protects the Warm Waste Pond at TRA. A USGS stream gaging station is located on the Big Lost River about 6000 feet southeast of the Warm Waste Pond. The USGS defined the discharge in the Big Lost River during a 300-year flood to be 5300 cubic feet per second ($ft^3\ s^{-1}$) (Hull 1989).

The INEEL facilities used large amounts of water from the Snake River Plain Aquifer. Site water usage averaged about 1×10^{10} gallons per year for 1979–1983 (Bowman et al. 1984). For releases of liquid wastes in the early years, discharges were made to wells, seepage basins or pits, or seepage ponds depending upon the facility. The adoption of a national policy to improve water quality and to reduce releases of liquid wastes to the environment at all Federal facilities in the late 1960s compelled sites like the INEEL to reexamine their liquid waste disposal practices, which had been in place for the previous 20 years or so (Nebeker and Lakey 1970; Dickey et al. 1972). As a result, methods of waste disposal, such as disposal wells or waste ponds, were reexamined and alternative disposal methods were proposed and cost estimates calculated.

The ICPP discharged liquid wastes via a 600-foot deep well, the bottom of which was 140 feet below the top of the water table. About 50 million gallons were discharged from the fuel storage basin to a seepage pit between 1954-1966. Liquid wastes from the NRF were discharged to a seepage pond that was backfilled with coarse gravel. At TAN, liquid effluents were discharged via four wells, although only one was used extensively. The ANL-W discharged liquids to a seepage pit. Liquid effluents from main facilities were generally monitored at the time of release. We compiled data of radioactivity levels and volumes discharged from key facilities at

the INEEL (see LiquidEffluents.xls). From reported discharges of liquid wastes, over 95% of liquid effluents over the years came from TRA and ICPP.

In 1970, tritium in liquid wastes was not considered a serious problem although some thought it prudent to know the sources (coolant or fuel), the production rates, and the discharge rates at points of release. The only measurements of tritium being made routinely by about 1970 were concentrations of tritium in liquid wastes being discharged to the ground at ICPP and TRA (Nebeker and Lakey 1970). Measured releases of tritium, which began in 1961 at ICPP and TRA, and total activity released from the major facilities to ponds or injection wells are tabulated in Excel spreadsheets. These values served as a basis for comparing quantities released among the facilities and for crosschecking other periodic reports or data tabulations that are reference in the next section. All facilities had liquid waste facilities and documented disposal methods and procedures (e.g. ANL-W 1973; Trojanowski 1974; Hogg et al. 1971a, 1971b).

To ensure that the liquid effluents released to the injection wells and onsite ponds did not lead to a potential complete exposure pathway for those offsite, we carefully reviewed the primary liquid waste disposal methods and procedures at ICPP and TRA, the greatest contributors to liquid effluent at the INEEL. Our evaluation concluded that the potential exposure pathway from liquid effluents discharged to the environment would be through groundwater. We evaluate that pathway in the Groundwater Pathway section of this Task Order 5 report.

Liquid Wastes from the ICPP

The ICPP was primarily designed to recover enriched uranium from spent reactor fuel elements. The uranium was recovered by a liquid-liquid solvent extraction process. The aqueous raffinate wastes containing the fission products from the extraction process were concentrated and stored in permanent underground tanks. Other liquids of large volume, containing low concentrations of radioactive materials, were "diluted and discharged to the subterranean environs by way of a 592 foot deep well." In 1954, a continuous liquid waste monitor was installed in the ICPP in a small underground building near the main process building (CPP 709) (King 1956). This monitoring system was designed to (1) continuously monitor the liquid wastes discharged to the well, (2) provide a record of the concentration of the beta-gamma activity in the waste stream, and (3) provide flow rate data from weir chambers. The monitor integrated the product of the flow rate and concentration over a given time period to give the total activity discharged. The system also integrated total volume of liquid discharged and collected a sample proportional to the flow rate for radiochemical analysis. Soon after installation, major changes were made in the detection units, which were replaced with scintillation type counters (King 1956).

Liquid waste streams were generated from all areas of the ICPP and they varied in volume and degree of contamination. The high level waste streams were sent to the WCF. Two intermediate and low-level waste streams were collected in the evaporator tank before discharge to the injection well. It was reported that the cell floor-drain collection system for low-level waste handled about 200,000 gallons per year and the process equipment waste collection system for intermediate level waste handled about 1 million gallons per year. Additional liquid wastes totaling about 15,000 gallons per year from other areas at the INEEL were handled in this system as well (Dickey et al. 1972).

Four waste streams entered the liquid waste monitoring system; three of the four streams carried radioactive discharges through three separate weir chambers, each of which could be

monitored separately. These three weir chambers spilled into the large main weir. The fourth waste stream came from a nonradioactive area, emptied into the main weir, and provided a dilution source. At the discharge end of the main weir, there was a 90-degree "V" notch that provided a method for metering the flow rate. In the 1950s the maximum flow rate measured was 1500 gallons per minute, calibrated in tens of gallons per minute (King 1956). A submerged pump removed a continuous sample of waste liquid just before discharge over the "V" notch and pumped it to the monitoring and sampling equipment. Two scintillation counters in lead shields monitored the liquids. In calibrations done with ^{137}Cs, the minimum concentration detectable was 0.0004 $\mu Ci\ mL^{-1}$ (400,000 picocuries per liter [$\rho Ci\ L^{-1}$]), and the maximum concentration detected was 0.05 $\mu Ci\ mL^{-1}$ (50 million $\rho Ci\ L^{-1}$). The wastes were then released to the ICPP injection well at a rate of about 1 million gallons per day in the 1970s. The 600-foot deep well consisted of a plastic pie inside a carbon steel shell and penetrated about 140 feet below the water table (Dickey et al. 1972).

Radioactive liquid releases from the ICPP, compiled from the quarterly effluent monitoring reports, summarized the effluent from the service waste system that was discharged monthly to the ICPP disposal well and later to the percolation pond. On February 9, 1984 flow to the ICPP injection well was officially terminated and the ICPP Percolation Pond came on line. However, flow to the injection well occurred periodically during the next 2 years. In 1984, flow to the injection well was reported on September 21, September 30, and November 14, and in 1985, flow to the well occurred six times when power or pump problems occurred. There were no emergency flows to the injection well in 1986, 1987, 1988, or 1989, and the injection well was permanently sealed in December 1989 (WINCO 1991).

After the injection well was shut down in 1984, the waste streams that carried the majority of the ICPP liquid pollutants were discharged through the East Side Service Waste (CPP-709) and West Side Service Waste (CPP-734). Both service waste streams entered the Percolation Pond in operation at the time (YDG-326 or YDG-327) (Krivanek 1988). During 1989 a new service waste system (CPP-796/-797) was placed in service (WINCO 1991).

The radionuclides listed in the Excel spreadsheet were measured; the listed radionuclides that do not have values were not found during that time period. In January 1983, there was a higher than normal reported release of ^{90}Sr resulting from an improper valve operation at CPP-603 (ENICO 1983a). Tritium contributed over 99% of the total radioactivity, measured in curies in the service waste effluent during most of the reporting periods. During the 1980s, monthly tritium releases to the ICPP injection well varied between 1 and 50 Ci, with releases of 100 Ci occurring in April 1981 and November and December 1982. With the startup of the NWCF and FAST, the total flow to the injection well was projected to increase from 460 million gallons per year to 1130 million gallons per year (Ritter 1981). There was an increase in tritium discharged in April 1983 due to operation of the liquid waste evaporator and the APS condenser, which was operated in May and June 1982 due to failure of a blower in the NWCF off-gas system (ENICO 1983b). In 1984, the volume of liquid waste effluent increased due to the new FAST facility and the Coal-Fired Steam Generating Facility (CFSGF) coming on-line during the year (WINCO 1985). Iodine-129 release trends were similar to those for tritium in the 1980s (ENICO 1983b).

Test Reactor Area (TRA) Liquid Wastes

TRA is a complex with three test reactors: MTR, ETR, and ATR. TRA used about 150 million gallons of water per month for reactor cooling, irrigation, and domestic use supplied from deep wells descending to the water table contained in the Snake River Plain Aquifer (about 450 feet below ground surface). Of the 150 million gallons, about 5% flowed to the desert because the capacity of the deep well pump exceeded the water requirements at TRA. About 10% was used untreated for miscellaneous cooling, irrigation, firewater, demineralizer regeneration, or domestic water use. Another 15% was demineralized for use in the reactor primary cooling systems for makeup of losses from the reactor experimental loops, for steam production, and for use at other TRA experimental facilities (Holcomb and Larrick 1974). Most of the water was used to replace water losses from the reactor secondary cooling systems due to evaporation and blow down. About one-third of the water used each month (50 million gallons) was discharged to seepage ponds and a disposal well as waste from the test reactor operations. There was concern though the late 1960s over the discharge of radionuclides, ^{90}Sr and ^{137}Cs, and chromium to wells even though there was no "legal limit" established at the time (Nebeker and Lakey 1970).

The TRA liquid wastes consisted of cold wastes; warm wastes (which contained a small amount of radioactivity but did not exceed the discharge limits for the time); hot wastes (which were too radioactive for immediate disposal to the groundwater); chemical wastes from the demineralizers and water softeners; and sanitary wastes. In 1971, the site reviewed and characterized all waste management procedures and provided characteristics, water disposal means, purge rate activity, and other parameters for intermediate-level liquid waste sources (Hogg et al. 1971a, 1971b). The report also provided the dimensions of the site's canal system; radioactive waste storage tanks volumes; location disposal methods; and cooling tower locations, sizes, volumes, disposal points, and monitoring protocols.

Evaluation of Buried Radioactive Solids

The INEEL has used several areas for solid radioactive waste materials disposal. The primary area has been the RWMC but other areas include the SL-1 Burial Ground (one trench and two pits 1600 feet east of old Stationary Low-Power Reactor No. 1 [SL-1] area), the ANL-W Solid Waste Storage Area (4 acres north of the EBR-II for scrap and solid wastes), and the ICPP Calcined Solid Waste Storage Area, where storage bins were put into service in 1963 for the storage of calcined waste. Originally at the RWMC and the SL-1 burial ground, "fission and activation products wastes" were buried directly in soil below ground level. While wastes containing transuraniuc and 233U activity above 10 nanocuries per gram were stored above the land surface "fire-resistant and watertight containers" (ERDA 1977).

The RWMC, opened July 8, 1952 with one trench, was the first location accepting radioactive wastes generated by INEEL operations. Over time the size of the RWMC increased from 13 acres in 1952, to 88 acres in 1988, and 144 acres by 1970. Solid waste sent from the Rocky Flats Plant in Colorado comprised a large fraction of the waste received at the RWMC. For example in 1969, approximately 250,000 cubic feet with a reported activity of over 35,000 Ci, originated from the Rocky Flats Plant in Colorado. The Rocky Flats waste was usually contaminated with plutonium isotopes and ^{241}Am but the monitoring of solid waste in the early years was minimal or nonexistent.

At the RWMC, burial or subsurface disposal in trenches and pits were the primary method of disposal. For disposal in the trenches through about 1973, the RWMC received *routine* or low-level radioactive waste in cardboard boxes sealed with masking tape. These were dumped into the trench, covered with soil, and compacted with a heavy steel plate dropped onto the waste. *Nonroutine* or high-level radioactive waste was placed in wooden boxes or 30-gallon metal cans. Concrete markers identified the end of each trench. The pits were opened in 1957 to handle large, bulky items, mainly from Rocky Flats. They were about 50–300 feet wide, 250–100 feet long and 5–15 feet deep. Large drums were hand-stacked and wooden crates were placed around the edge of the pit and the waste was periodically covered with soil. By late 1963, the Rocky Flats waste volume had increased considerably and the waste was simply dumped into the pits. This random dumping continued until 1969 (Smith 1981). Concrete markers were put in place and identified the center of each pit.

Later, the Transuranic Storage Area (TSA), Transuranic Disposal Area (TDA), and Intermediate Level Transuranic Storage Facility (ILTSF) were developed. The TSA was designed for "interim storage" for 20 years. Here the waste containers were stacked, covered with plywood, nylon reinforced polyvinyl, and soil. The TSA was used from November 1970 through October 1975, and TSA-2 received waste from September 1975 through June 1980, and had an air support weather shield. Next, the ILTSF was constructed in late 1975 to receive waste that required special handling but was not high-level waste. The IFTSF was below-grade storage in carbon-steel pipe vaults 12 and 16 inches in diameter. The vault was embedded in compacted soil 30 feet wide, 350 feet long, and 5 feet high, and extend 4 inches above a thick asphalt pad. Figures 13 and 14 give a comparative view of the types of wastes located at the RWMC and the source of the solid wastes at the INEEL (Osloond 1970; Smith 1981).

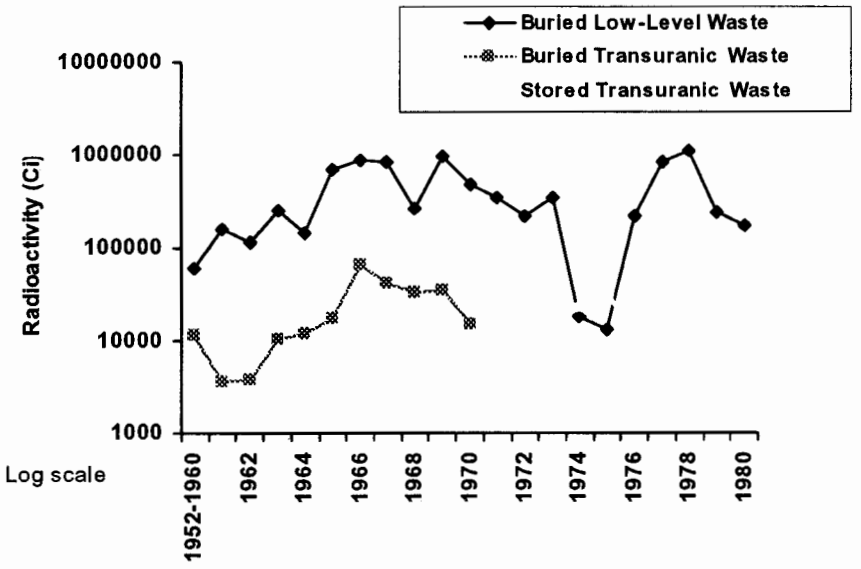

Figure 13. The categories of solid wastes located at the Radioactive Waste Management complex at the INEEL. In the early 1970s, transuranic wastes were stored in above ground facilities and were no longer buried.

Several major events at he IEEL and specifically at the RWMC led to modifications in the procedures and burial practices as the site. After the SL-1 accident in January 1961, most of the waste was buried at a special location, the SL-1 Burial Ground, about ¼ mile from the reactor location. However, some of the waste was put into Pit 1 at the RWMC, which was reopened in October 1961 to accept this waste. In February 1962, the RWMC suffered a severe flood when 2 inches of rain and 8 inches of snow fell in three days, which was followed by a warming trend. With the upper foot of ground frozen, extensive runoff occurred into open pits and trenches containing boxes and barrels of radioactive waste. Pits 2 and 3 and Tranches 24 and 25 were particularly hard hit and resulted in waste floating in the flood water. Extensive radiation surveys were done and water samples collected from surrounding pits and ponds, and much of these data are available (Smith 1981). A diversion drainage system was constructed around the perimeter of the Burial Ground as a result of this flood. In January 1969, another flood occurred at the RWMC because snowdrifts blocked the existing drainage system. Water entered trenches 4 and 48, filled Pit 10, and partially filled Pit 9. New larger dikes and ditches were constructed in response to this flood.

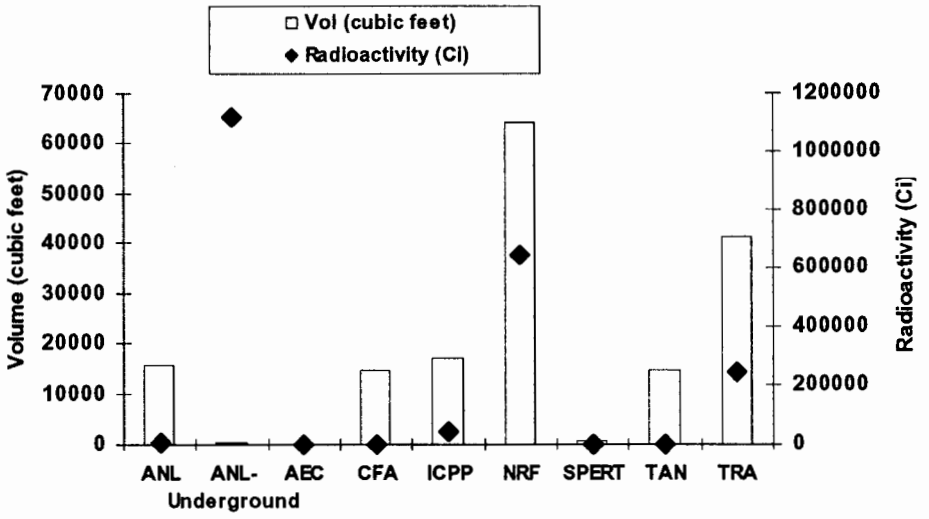

Figure 14. Sources of solid waste at the INEEL in 1969. The columns represent the volume of waste (left axis) and the filled diamond shapes represent the level of radioactivity in the waste (right axis). The Naval Reactor Facility and Test Reactor Area contributed large volumes of solid waste while waste at the ANL-W underground facility has the highest activity, based on data from Osloond (1970).

The occurrence of several fires at the RWMC led to changes in disposal procedures and safety measures, too. Fires in September 1966 in Trench 42 occurred when alkali metals in waste interacted with low-level radioactive waste in the open air when there was a delay in compacting and covering the waste. At that time, compaction and burial were carried out when necessary and not on a regular basis. A subsequent directive in October 1966 specified that waste be compacted and covered with soil on a weekly basis. There was another fire in June 1970 fire in an above ground storage area

The ICPP disposed of its radioactive solid waste at the RWMC and the Radioactive Shipping Coordinator (ENICO 1983a) kept records of these shipments. While the monthly volume of solid radioactive waste varied between 50 and 200 m^3, high activity levels in the waste occurred in December 1981 (20,000 Ci); May, June, and July 1982 (~50,000 Ci); and October 1983 (1000 Ci). It was noted that the high volume of waste in the third quarter of 1983 (1900 m^3) was due to shipping large volumes of previously stored materials from construction and demolition activities (WINCO 1984).

The potential exposure pathway of leakage from underground storage of solid waste at the radioactive waste areas at the INEEL would be through groundwater. We evaluate that groundwater pathway in a later section of the report.

Screening Methodology For Routine Releases

Radionuclides released routinely from the INEEL are those expected from reactor and reprocessing operations. However, there can be differences in the amounts released to air and water, the half-life, the behavior of the material in the environment, and biological uptake so the greatest amount of radioactivity released does not necessarily translate into the highest dose to a nearby person. It is impossible to give equal attention to each radionuclide in the early stages of dose reconstruction. The screening process helps to focus the research efforts so that resources will be allocated to the prioritized radionuclides, time periods, and events that contribute most to doses to those nearby or in surrounding communities.

NCRP Methodology

The relative importance of releases of radionuclides to the environment depends upon the quantities released, differences in the potential for nuclide concentration in the environment, and the relative toxicity of the radionuclides, as measured by established dose conversion factors. The method used to screen radioactive contaminants potentially released from the Site to the environment was one developed by the NCRP (NCRP 1996). The methods and reports have been extensively reviewed and are widely accepted. The method uses a phased approach, from simple calculations using very cautious assumptions to a more complex evaluation using site-specific data, when available. Cautious or conservative calculations that overestimate the doses from radionuclides produce a ranking of important radionuclides in terms of radiation dose to people potentially exposed to them. The radionuclides ranked low on the basis of the screening calculation are not likely to be important.

The NCRP screening methodology is a valuable tool because it provides a compilation of effective dose factors and screening factors for exposure pathways of more than 800 radionuclides and generic environmental transport parameters, including uptake, bioaccumulation, and environmental transfer factors. The information for each radionuclide is encapsulated in the total screening factor, which is the sum of committed effective doses received from inhalation; plume immersion; external irradiation from ground contamination; and ingestion of soil, vegetables, milk, or meat assumed to be locally produced during 1 year for a unit concentration of radioactivity in air. Screening factors for a radionuclide are also provided by pathway so the dominant exposure pathway for that radionuclide can be evaluated. The screening factors assume an average annual air concentration and a 30-year buildup time to account for

accumulation in the environment. The resulting screening value or dose applies to a period of 50 years following the release. The total screening factors are most appropriate to use for evaluating offsite exposure at a potential residence location because they include contributions from all pathways. For screening routine releases at the onsite location (Highway 20), we used only the inhalation and plume immersion screening factors reported by NCRP (1996) The inhalation and plume immersion screening factors are more appropriate for individuals who may have been on or near the site or passing through the site for a portion of the year. For our offsite exposure assessment for routine releases, we assumed a location at Atomic City, 20 km from the ICPP and TRA, and used the total screening factors for the radionuclides.

The first step in applying the screening methods to atmospheric releases from the INEEL was assessing the releases of particular radionuclides from the facilities at the INEEL during their operational history, as we discussed in previous sections of the report. We then applied conservative and simple transport models to the releases and incorporated human consumption rates and usage factors that were quite cautious or conservative and tended to overestimate the parameters used in the screening calculations. Table 3 provides examples of some individual usage factors in the screening and illustrates the use of conservative values.

Table 3. Annual Individual Values Used in the NCRP Screening Models[a]

Exposure pathway	Selected parameters	NCRP value
Inhalation pathway	Breathing rate	$8000\ m^3\ y^{-1}$
	Resuspension factor	$2 \times 10^{-8}\ m^{-1}$
External exposure	To contaminated ground surface (assume exposed most of the year)	$8000\ h\ y^{-1}$
Ingestion pathway	Vegetable, fruits, grains (assume root uptake and soil adhesion)	$100\ kg\ y^{-1}$
	Water (assume drinking water from area)	$800\ L\ y^{-1}$ or $2.5\ qt\ d^{-1}$
	Milk (assume no milk from other areas)	$300\ L\ y^{-1}$ or $0.82\ qt\ d^{-1}$
	Soil	$0.25\ g\ d^{-1}$

[a]From NCRP 1996.

The NCRP approach considers environmental transport mechanisms, exposure pathways, and radiation dosimetry in a few simple steps. In the first step, the concentration of the radionuclide in the environment was calculated by using environmental transport screening models and the release quantity from the facility. The environmental concentration was then multiplied by a screening factor for that particular radionuclide to obtain a screening value that was compared with screening values for other radionuclides released from the INEEL. For screening air releases, a simplified ground-level, centerline Gaussian plume atmospheric dispersion model was used (se the Episodic Release Evaluation Methodology section for more details on this model).This model assumed that a flat terrain, similar to the INEEL area. If we assumed a ground level effluent release as a conservative approach, then the atmospheric concentration, C, of a particular radionuclides was calculated as follows:

$$C = \frac{f Q}{\pi u \delta_y \delta_z} \quad (1)$$

where

C =is the atmospheric concentration of the radionuclide (Bq m^{-3})
Q =is the release rate from the facility, or source term (Bq s^{-1})
f =is the fraction of time the wind blows toward the person (dimensionless).
u = mean wind speed at height H (m s^{-1})
δ_y = horizontal dispersion coefficient at coordinate x (m)
δ_z = vertical dispersion coefficient at coordinate (m)

For screening annual average atmospheric concentrations, it was assumed that the atmospheric stability was neutral (Pasquill category D). As a result, the horizontal and vertical dispersion coefficients can be defined as follows (NCRP 1996):

$$\delta_y = \frac{0.08\,x}{\sqrt{1 + 0.0001\,x}} \quad (2)$$

$$\delta_z = \frac{0.06\,x}{\sqrt{1 + 0.0015\,x}} \quad (3)$$

where

x = distance downwind from the source (m)

The closest onsite location routinely accessible to members of the public was Highway 20, which intersects the site about 6 km south of the ICPP and TRA, the two facilities routinely contributing the highest activities (Figure 1). For a distance of 6 km, the horizontal dispersion coefficient (δ_y) was 379 m and the vertical dispersion coefficient (δ_z) was 114 m. We used the Highway 20 location (6 km) as our onsite exposure point. For offsite exposure, we assumed a location in Atomic City, 20 km from the site. For a distance of 20 km, the horizontal dispersion coefficient (δ_y) was 924 m and the vertical dispersion coefficient (δ_z) was 216 m. These distances of 6 km (onsite) and 20 km (offsite) from the point of release to the closest person ensured a conservative approach because it assumed that location for the entire year. The release rate, Q, for each radionuclide is based on estimates of the amount released in a 1-year period from the facility. For f, we assumed that the wind blows 100% of the time toward the potentially exposed individual, so f =1.

The Site compiled wind speed data since 1973 when 26 meteorological stations were established around the INEEL. During that time period, the annual average wind speeds ranged from about 2.5 m s^{-1} in 1985 and 1989 to over 7.5 m s^{-1} in 1973, 1977, and 1988 (DOE 1991a). To ensure a conservative screening approach, we assumed a wind speed, u, of 2.5 m s^{-1}.

Releases from Facilities, or Source Terms

An important component of this process, and of Equation (1), is the source term, or estimates of releases of individual radionuclides from INEEL facilities during all years of operation. Because the screening factors assume an average annual air concentration, we compiled annual

airborne release estimates from the facilities for each year from 1952 through 1992. For a screening process, we used information from several sources. For the most part, only Site personnel monitored and recorded effluent data at points of release onsite at the INEEL. Because the majority of historic monitoring and record keeping came from the Site, we must rely on available Site records for our screening efforts. In addition to Site monitoring and process records, however, we can also draw on the basic chemistry and nuclear physics of the reactor and chemical plant operations the INEEL. The process engineering for the chemical plant areas and the nuclear reactors are quite well understood. Therefore, it is possible to estimate the types and relative quantities of materials that might be expected from a particular process or reactor operation run. This information was useful for radionuclides that were not measured during particular years of operations, or from particular facilities.

For annual releases to air for early years of operation, the Historic Dose Evaluation (DOE 1991a) data proved to be the most complete, because adjustments had been made to the RWMIS data that were the basis for the Historic Dose Evaluation source term data. The focus of (DOE (1991a) was the calculation of doses based on historic atmospheric effluent release data, atmospheric dispersion calculations that reflected the meteorological conditions of the INEEL area, and internal and external dose conversion factors (DOE 1991a, Appendix C). For the Task Order 5 screening calculations, we focused on the operational atmospheric source term data from the DOE (1991a) report. These release data had included a series of adjustments to the RWMIS database, especially for earlier years, to ensure that the reported mixture of radionuclides released to air were as realistic as possible based on past measurement techniques and basic decay chain physics. DOE's review revealed technical problems with the RWMIS data from 1962 to 1968. To rectify these difficulties, release data from the ICPP and the TRA were extensively revised. Some of the key adjustments were

- DOE (1991a) assumed the equilibrium relationship between parent and daughter radionuclides whenever it would have a significant effect on the calculations of external dose to an offsite individual. In particular the parent and daughter relationships for $^{89}Kr/^{89}Rb$, $^{92}Sr/^{92}Y$, $^{105}Ru/^{105}Rh$, $^{127}Sb/^{127}Te$, $^{131}Te/^{131}I$, $^{134}Te/^{134}I$, $^{138}Xe/^{138}Cs$, and $^{142}Ba/^{142}La$ were reviewed, assumed to be in equilibrium, and each radionuclide treated independently (DOE 1991a, Appendix A)
- Releases of the noble fission gases, krypton and xenon, from the TRA from 1952 through 1968 were recalculated because of anomalies in the reported RWMIS releases of ^{137}Xe and ^{138}Xe, and ^{88}Kr and ^{89}Kr for that period. While the Radiological Safety Analaysis Computer Program (RSAC)-4 computer code indicated that the release quantities of ^{88}Kr and ^{138}Xe should have been greater than the quantities for ^{89}Kr and ^{137}Xe, respectively, the RWMIS reported just the opposite: the release of ^{137}Xe, not the release of ^{138}Xe.
    - To correct this anomaly, the revised estimates reported in DOE (1991a) for the xenon and krypton gases from TRA were determined from the reported total annual airborne effluent activity, based on the facility cycle reports for the years 1960 through 1963, and the reported percentages of ^{41}Ar, gaseous activity, and particulate activity.
    - These percentages were applied to the annual TRA airborne effluent for 1952 through 1968. To further breakdown the gaseous and particulate components

into individual radionuclides, the proportions applicable to ATR based on the isotopic composition for 1987 airborne effluent were used.

- DOE (1991a) assumed all "gross beta" activity was ^{90}Sr, and all "gross alpha" activity was composed of ^{238}Pu, ^{239}Pu, and ^{240}Pu in the same ratio as released from the ICPP during a 13-year period from 1974-1986 when specific plutonium analyses were done. This ratio was applied to the 1969-1976 period. For 1964-1968, no alpha emissions were reported, but it was assumed that plutonium releases did occur during those years. In this case, the DOE (1991a) used the ratio of ^{90}Sr emissions to total plutonium emissions for 1969 through 1974, which was about 400. To estimate plutonium activity for 1967 and 1968, they divided the ^{90}Sr emissions by this ratio and applied the $^{238}Pu/^{239,240}Pu$ ratio calculated from the 1974 to 1986 period.

We carefully reviewed the rationale for the adjustments made to the RWMIS data and determined that they were technically defensible and soundly based.

Reported annual data were tabulated in Excel spreadsheets for specific radionuclides released from facilities that contributed to the largest total releases to air, ANL-W, ICPP, and TRA (FacilityAirReleases.xls). Annual release estimates were compared with monthly release data for certain time periods reported at the site for specific facilities to corroborate reported values. As we compiled release data for specific radionuclides, we noted changes in analytical methods, instrumentation, new programs, or episodic events that affected the radionuclides monitored, or the way in which the monitoring and analysis were done. For example, on August 21, 1984, the Analytical Chemistry Group at the ICPP began using a new computerized gamma-detecting instrument on all radiological samples. It appears that this instrumentation improved the analytical results. As we moved forward in time this type of improvement led to increasing confidence in the measurement techniques, analytical methods, and reporting capabilities of the Site compared to early years of operation when special radiological analyses were not done in may cases.

Where data were not reported, we used a conservative approach to estimate releases of radionuclides for years when the release of a specific radionuclide was expected. At times, we used more recent data, with appropriate modifications, to fill in data gaps during earlier time periods. Annual releases from the INEEL provided the basis for the screening calculations. The Excel spreadsheets that contain these data note how estimates were determined if releases were not directly reported in a site document (see Offsite Air Screening.xls and Onsite Air Screening.xls). For screening, we used a conservative approach to ensure that we have not overlooked radionuclides or time periods when measurements were not made. In some cases, the assumptions may appear to be extremely conservative. However, if our screening results show that a particular radionuclide was not an important contributor to dose, even with very conservative assumptions, then we can feel confident that it will not be important under more realistic conditions. The same logic applies to assumptions based on reactor or processing activities. The relative quantities and importance of activation or fission products produced would depend on a number of factors in effect at that time, such as the composition of materials in the reactor, the power level, flux conditions, and previous operating parameters. For screening analysis, we selected conservative ratios or parameters so that our results would clearly show whether a radionuclide may be an important contributor to dose. Again if results indicated that a radionuclide was not important with conservative assumptions, we were assured that it would not

be an important contributor to dose with more realistic assumptions. Here we summarize the approach used for several radionuclides as examples of our conservative approach.

Carbon-14, which is produced in reactors as a result of neutron absorption by nitrogen, carbon, or oxygen present as components of air, coolant, moderator, structural materials, fuel, or impurities, was reported for years after 1978. To estimate releases of ^{14}C for years before 1978, we calculated the $^{41}Ar/^{14}C$ ratio of releases to air from 1978 through 1992. The ratios varied from about 450 to 26,000 with a geometric mean of 3200. To ensure a conservative approach we used the lowest ratio (450) and divided the measured annual releases of ^{41}Ar by this ratio to maximize releases of ^{14}C. Table 4shows the releases and ratio of $^{41}Ar/^{14}C$.

Table 4. Calculated $^{41}Ar/^{14}C$ ratios for Years When Both Were Measured [a]

Year	Measured releases (Ci)		$^{41}Ar/^{14}C$
	^{41}Ar	^{14}C	
1978	3800	9.10	420
1979	3400	1.1	3100
1980	2200	4.3	510
1981	2500	1.6	1560
1982	2500	0.29	8620
1983	2300	0.23	10000
1984	1800	0.33	5450
1985	2100	0.70	3000
1986	1800	0.61	2950
1987	2500	4	625
1988	2100	2.7	780
1989	1400	0.21	6700
1990	3300	0.28	12000
1991	2900	0.11	26400
1992	2500	0.14	18000

[a] For a conservative approach to estimating ^{14}C releases for years before 1978, we used the lowest ratio of $^{41}Ar/^{14}C$ (420) based on measured releases from 1978–1992 and applied it to measured annual releases of ^{41}Ar before 1978 to estimate ^{14}C releases for those years.

The isotope ^{99}Tc is a fission product of both uranium and plutonium and has a mass yield comparable to ^{90}Sr. It has a longer half-life than ^{90}Sr and low specific activity (0.017 Ci g^{-1}), so the activity of ^{99}Tc will be lower than that of ^{90}Sr. Technetium-99 emits a weak beta and no gamma, so it was not considered an important radionuclide and no measurements of ^{99}Tc were made at the INEEL. We estimated atmospheric releases of ^{90}Tc using the fission product ratios (FPRs) of $^{90}Sr/^{99}Tc$ of 5300 and $^{137}Cs/^{99}Tc$ of 7600 and applying those ratios to measured releases of ^{90}Sr and ^{137}Cs, respectively (Till 1984). When these ratios were applied to measured releases of ^{90}Sr and ^{137}Cs from the INEEL, two sets of annual release estimates were obtained for ^{99}Tc. We selected the highest release estimate for each year to ensure a conservative screening analysis. Table 5 summarizes the results of this assessment and, the last column, shows the annual releases of ^{99}Tc that we used for the screening assessment. Based on this method, the total releases of ^{99}Tc

from the INEEL from 1952 through 1992 were estimated to be 0.15 Ci, compared to 340 Ci of ^{90}Sr and 880 Ci of ^{137}Cs.

Table 5. Method for Estimating ^{99}Tc Releases to Air from the INEEL Based on Fission Product Ratios (FPR) for ^{90}Sr and ^{137}Cs[a]

Year	Reported ^{90}Sr releases (Ci)	^{99}Tc releases (Ci) based on FPR of 5300 for $^{90}Sr/^{99}Tc$	Reported ^{137}Cs releases (Ci)	^{99}Tc releases (Ci) based on FPR of 7600 for $^{137}Cs/^{99}Tc$	Estimated ^{99}Tc releases (Ci) [b]
1952	1.6E+00	3.0E-04	3.8E-01	4.9E-05	3.0E-04
1953	8.4E+00	1.6E-03	5.8E+00	7.7E-04	1.6E-03
1954	1.5E+01	2.8E-03	1.3E+01	1.7E-03	2.8E-03
1955	2.1E+01	4.0E-03	1.9E+01	2.4E-03	4.0E-03
1956	2.2E+01	4.1E-03	2.1E+01	2.7E-03	4.1E-03
1957	3.6E+01	6.7E-03	3.6E+01	4.7E-03	6.7E-03
1958	5.1E+01	9.6E-03	5.1E+00	6.7E-04	9.6E-03
1959	4.3E+01	8.0E-03	4.2E+01	5.5E-03	8.0E-03
1960	2.6E+00	4.9E-04	9.6E-01	1.3E-04	4.9E-04
1961	3.2E+00	5.9E-04	7.7E-01	1.0E-04	5.9E-04
1962	4.7E+00	8.8E-04	2.2E+00	2.9E-04	8.8E-04
1963	3.4E+01	6.5E-03	3.1E+00	4.0E-04	6.5E-03
1964	8.8E+00	1.7E-03	4.8E+00	6.3E-04	1.7E-03
1965	3.4E+01	6.3E-03	1.4E+01	1.8E-03	6.3E-03
1966	9.6E+00	1.8E-03	5.3E+00	7.0E-04	1.8E-03
1967	2.6E+00	4.9E-04	1.4E+00	1.8E-04	4.9E-04
1968	1.5E+01	2.8E-03	6.7E+02	8.8E-02	8.8E-02
1969	4.4E+00	8.3E-04	4.3E+00	5.7E-04	8.3E-04
1970	3.3E+00	6.2E-04	2.3E+00	3.0E-04	6.2E-04
1971	1.4E+01	2.6E-03	1.5E+01	2.0E-03	2.6E-03

[a] Although releases and estimates were done for all years, we show only 20 years here.

[b] Values in this column were used in the screening assessment.

For some radionuclides with short half-lives, measurements were reported only for years of the RaLa runs when short-cooled fuel was processed. For example ^{132}Te and its decay product, ^{132}I, were reported for 1956–1963 and 1967.

Releases of ^{13}N were reported only in 1972. For other years, we assumed releases of ^{13}N twice that reported in 1972. Similarly, releases of ^{76}As were reported only in 1975; for other years we assumed releases of ^{76}As as twice that reported in 1975. Because ^{129}I was not reported until 1979, the ^{129}I releases reported in DOE (1991a) were based on the amount of ^{129}I in the fuel that was processed and on studies that indicated that most was released during waste calcinations at the ICPP. About one-third of the ^{129}I released from the ICPP was elemental (I_2) and two-thirds in the organic form, which is less important for dose to local residents because it does not deposit as readily on vegetation. For our conservative screening, we assumed all ^{129}I was in the elemental form and used the screening factor accordingly.

For ^{125}Sb, which is formed through activation of ^{124}Sb and electron capture and was released primarily from the ICPP, releases were reported from 1965–1992 (DOE 1991a). For 1952–1964, we assumed releases of ^{125}Sb based on the arithmetic mean of measured releases. To ensure a conservative screening method, we doubled that value as a conservative estimate for early years of operation. The arithmetic mean for reported releases from 1965–1992 was 1.44 Ci × 2 = 2.9 Ci, which is the assumed annual release for the early years. For ^{82}Br, releases to air were reported from 1973–1990 from the EBR-II at ANL-W. Because EBR-II came online in 1961, we assumed releases from 1961–1972 as twice the average annual release for 1973–1990 to ensure a conservative screening approach.

Cesium-134 releases came primarily from the ICPP.This isotope is produced by neutron activation of the stable fission product ^{133}Cs. Releases were reported for from 1952–1981. The arithmetic mean of ^{134}Cs releases for 1952–1981 was 2.9 Ci. We assumed releases from 1982–1992 as twice the average annual release for 1952–1981(5.8 Ci) to ensure a conservative screening approach.

Releases of the activation product, ^{54}Mn, were reported from the INEEL for 1965 and 1973-1975. For release estimates during other years, we assumed annual releases based on the ratio of releases of ^{54}Mn to another activation product, ^{60}Co, which was released from routine reactor operations at a similar rate (Eichholz 1983). We calculated a ratio of 3 (ratio ranged from 3 to 20) of ^{60}Co to ^{54}Mn for years when both were reported and applied that ratio to years when ^{54}Mn was not reported.

Releases of the activation product ^{60}Co were reported from 1952-1974 and in 1987. Presumably, no ^{60}Co releases were reported after 1974 because the APS had been installed in the ICPP and was operational by 1975. For conservative screening, we assumed annual releases from 1975 through 1992 as the average of releases from 1970 through 1974 (0.0035 Ci). When annual releases of other activation products were not reported, we related releases to that of ^{60}Co. We calculated the annual releases of ^{58}Co based on 10 times the ratio of measured releases of ^{58}Co to ^{60}Co in 1973 (the only year that annual releases of ^{58}Co were reported), and we applied that ratio to all other years.

Plutonium consists primarily of approximately 93.8% ^{239}Pu, 5.8% ^{240}Pu, and 0.36% ^{241}Pu. Americium-241 is a decay product of ^{241}Pu, which is produced from neutron capture by ^{239}Pu in reactors and can be released during fuel processing. The decay products of ^{241}Am emit gamma radiation, but ^{241}Am was not monitored in airborne effluents at the INEEL. For our conservative screening, we assumed annual releases of ^{241}Am equal to 0.36 % of reported $^{239,240}Pu$ releases (the percentage of plutonium that would be ^{241}Pu.

Results of Screening Routine Releases

Radionuclides released to air from the INEEL facilities were conservatively[6] assessed in several ways to determine the radionuclides that contributed most significantly to the screening

[6] By conservative, we mean that we have utilized selected parameter values (e.g., gross activity and fractional release estimates), dispersion calculation methodology, and assumed exposure locations for each calculation that forced our estimated screening values to likely be significantly higher than actual dose estimates to ensure that we have not underestimated the potential impact of any release.

value[7] for onsite and offsite exposures for all years of operations and for individual years. The screening process also identified the individual years that were most important for onsite and offsite exposures, the radionuclides that contributed the highest percentage to the screening value for individual years, and the facilities that contributed most to the offsite screening value. Specifically, we present the results of the routine screening in the following ways:

- Offsite exposure (in Atomic City) to radionuclides released from the INEEL for all pathways for individual years from 1952–1992
- Onsite exposure (Highway 20) to radionuclides released from the INEEL for the inhalation and plume immersion pathways for individual years from 1952–1992
- Offsite exposure (in Atomic City) to radionuclides released from key facilities (ICPP, TRA, ANL-W) for high release years (1952–1964). During this time, releases occurred when effluents were not treated, filtered or monitored to the same extent as in later years.

The output from each screening calculation was a list of screening values for individual radionuclides for each year that provided the basis for prioritizing the radionuclides. We summed the screening values and calculated the relative contribution of each radionuclide to the total screening value for that year for all pathways of exposure for offsite exposures (or for the inhalation and plume immersion pathways for onsite exposures). All input release estimates, computations, and results of the NCRP screening methodology were compiled in Excel spreadsheets for each screening scenario (see Offsite Air Screening.xls and Onsite air Screening.xls). Based on these screening results, we prioritized the radionuclides, the years, and the facilities that contributed significantly to the screening values for routine releases.

Establishing Relative Importance of Radionuclides

Figure 15 is a composite of three graphics that shows the results of the offsite screening all radionuclides, organized by atomic number, for all years from 1952–1992 considering all pathways (inhalation; plume immersion; ground contamination; and ingestion of meat, vegetables, and milk from areas exposed to contaminants). When considering all radionuclides released from the INEEL for all years, this figure shows that ^{137}Cs, ^{131}I and ^{90}Sr contributed approximately 50%, 20% and 15% of the total screening value, respectively. Radionuclides contributing between 1 and 10% to the total screening value included ^{41}Ar, ^{106}Ru, ^{134}Cs, and ^{144}Ce.

[7] The screening value is reported in units of dose (millirem or sievert), but does not represent a "true dose" because of many conservative assumptions used in the screening analysis for both routine and episodic releases.

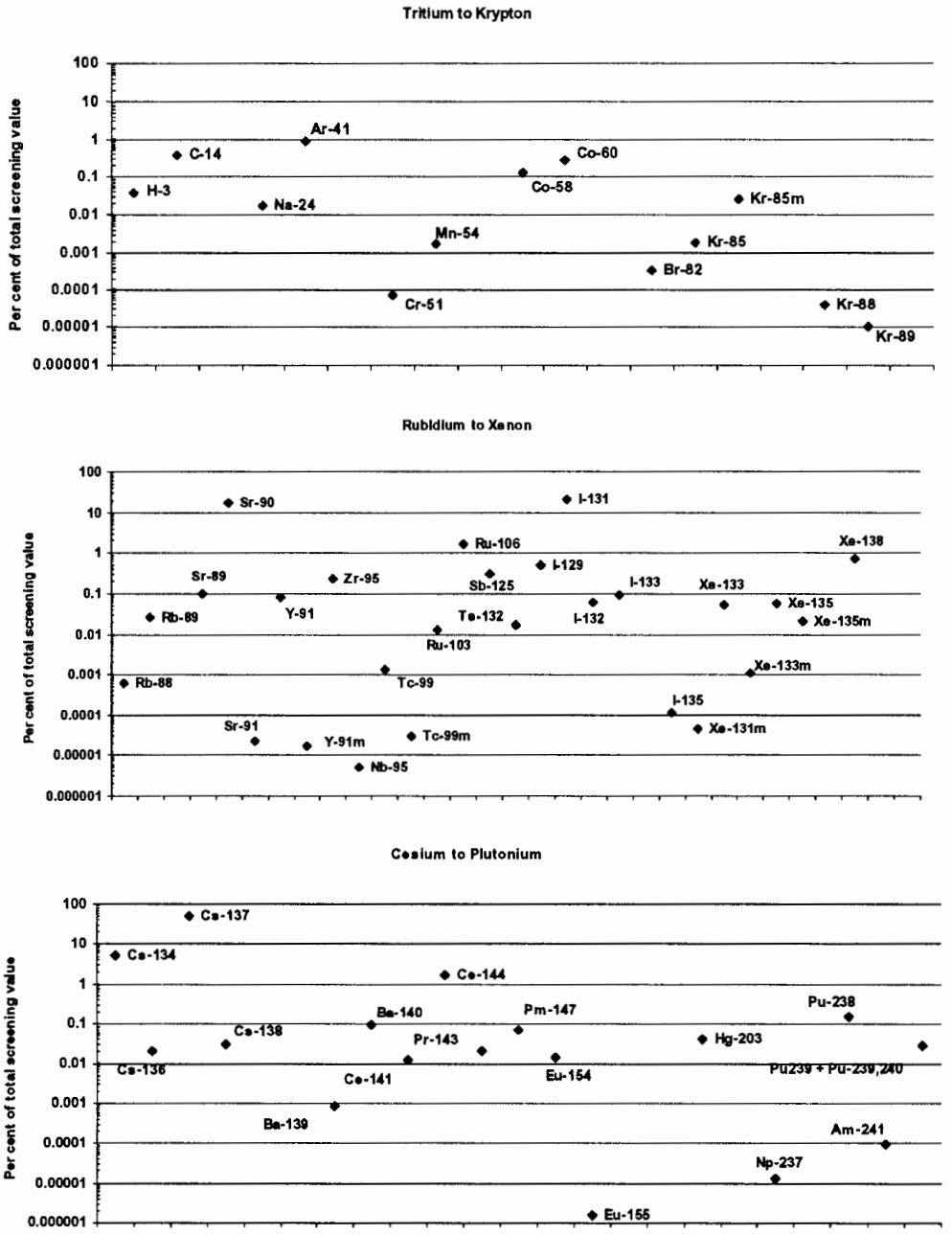

Figure 15. Results of offsite screening radionuclides that were released from routine operations at the INEEL for 1952–1992 considering all pathways of exposure in the NCRP methodology. Three graphics were used to show the results more clearly. Radionuclides are ordered by mass number from tritium to plutonium. The left axis is a log scale and shows the relative contribution of each radionuclide to the total screening value, which is 100%. For example, ^{155}Eu contributes 0.000001% (lower graphic), ^{99}Tc contributes 0.001% (middle graphic), ^{41}Ar contributes 1% (top graphic), while ^{137}Cs contributes about 50%, ^{131}I contributes over 20%, and ^{90}Sr contributes about 15% to the total screening value for all years.

The results for screening releases for individual years, the radionuclides contributing the most to the screening value varied somewhat. Figure 16 is a similar graphic that shows the results of onsite screening considering only the inhalation and plume immersion pathways. For onsite exposures where plume immersion and inhalation were the essential pathways, releases of ^{41}Ar and ^{138}Xe were the largest contributors to the screening value.

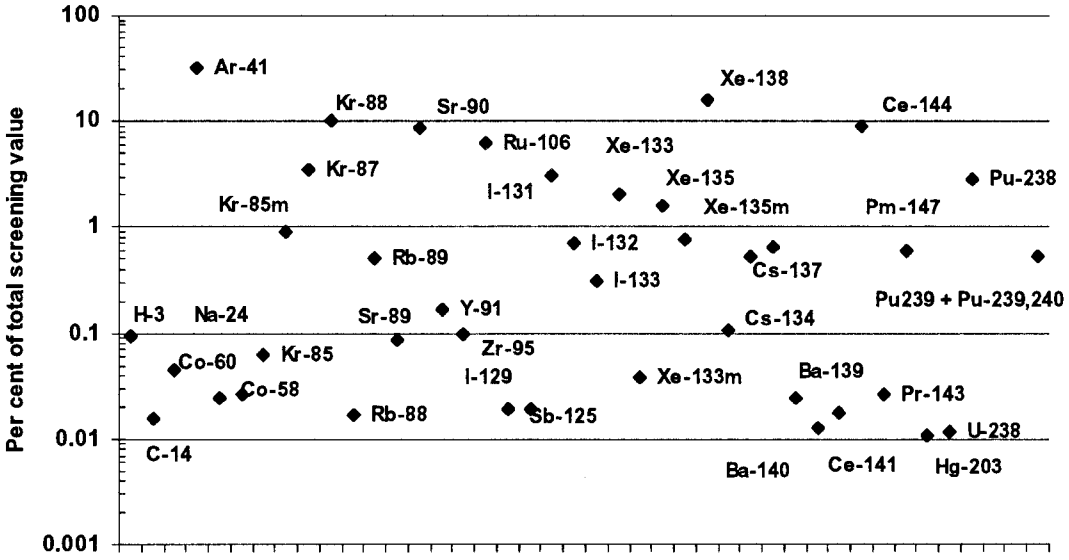

Figure 16. Results of onsite screening of the radionuclides released from routine operations at the INEEL for 1952–1992 considering the inhalation and plume immersion pathways of exposure in the NCRP methodology. Radionuclides are ordered by mass number and the figure shows only those radionuclides contributing greater than 0.01% to the screening value. The left axis is a log scale and shows the relative contribution of each radionuclide to the total screening value, which is 100%. For onsite screening, the noble gases, ^{41}Ar, krypton and xenon isotopes, were major contributors to the screening value.

Results from selected screening runs for offsite and onsite exposure for all years of operations are shown in Tables 6 and 7, respectively. The tables list the radionuclides that contributed over 99% of the total screening value. The radionuclides that are highlighted, collectively contributed more than 95% of the total screening value. For the offsite location, the radionuclides, ^{137}Cs, ^{131}I, and ^{90}Sr, were the largest contributors to the total screening value overall. For the onsite location, there was a greater variety of radionuclides contributing to the screening value; most are the krypton and xenon isotopes along with ^{41}Ar. Still, ^{131}I and ^{90}Sr were among the main radionuclides that contributed to the onsite screening value.

Table 6. Screening Results for Offsite Exposure to Average Radionuclide Releases for All Pathways for1952-1992 [a]

Radionuclide	Half-life	Total releases 1952-1992 (Ci)	Air screening value (Sv)	Air screening value (mrem)	Percent of total screening value
^{137}Cs	30.1 y	880	0.0036	360	49
^{131}I	8.04 d	2850	0.0015	150	21
^{90}Sr	27.7 y	340	0.0012	120	17
^{134}Cs	2.06 y	150	0.0038	380	5
^{144}Ce	284 d	1200	0.00017	17	2
^{106}Ru	1 y	690	0.00014	14	2
^{41}Ar	1.8 h	2,340,000	0.000066	6.6	0.9
^{138}Xe	17.5 min	1,260,000	0.000052	5.2	0.7
^{129}I	15700000 y	3.5	0.000039	3.9	0.5
^{14}C	5730 y	5700	0.000027	2.7	0.4
^{125}Sb	2.7 y	77	0.000023	2.3	0.3
^{60}Co	5.27 y	6.6	0.000022	2.2	0.3
^{95}Zr	65.5 d	220	0.000017	1.7	0.2
^{238}Pu	86.4 y	0.6	0.000012	1.2	0.1

[a] Of the radionuclides evaluated, these contributed more than 99% to the total screening value; the highlighted radionuclides represent more than 95% of the screening value for offsite exposure from all pathways.

Table 7. Screening Results for Onsite Exposure to Average Radionuclide Releases forInhalaltion and Plume Immersion Pathways for1952-1992 [a]

Radionuclide	Half-life	Total releases 1952-1992 (Ci)	Screening value (Sv)	Screening value (mrem)	Per cent of total screening value
^{41}Ar	1.8 h	2.34E+06	3.0E-04	30	30
^{138}Xe	17.5 min	1. 26E+06	1.5E-04	15	15
^{144}Ce	284 d	1.21E+03	1.2E-04	12	12
^{88}Kr	2.86 hr	3.91E+05	9.8E-05	9.8	9
^{90}Sr	27.7 y	3.40E+02	8.2E-05	8.2	8
^{106}Ru	1 y	6.94E+02	6.5E-05	6.5	6
^{87}Kr	1.27 hr	3.97E+05	3.6E-05	3.6	3
^{238}Pu	86.4 y	6.37E-01	3.1E-05	3.1	3
^{131}I	8.04 d	2.85E+03	3.0E-05	3.0	3
^{147}Pm	2.62 y	8.10E+02	2.1E-05	2.1	2
^{133}Xe	5.25 d	5.27E+06	2.1E-05	2.1	2
^{135}Xe	9.1 h	5.88E+05	1.6E-05	1.6	1.5
^{132}I	2.28 h	1.99E+04	1.5E-05	1.5	1.4
^{85m}Kr	4.48 hr	5.08E+05	1.0E-05	1.0	0.9
^{135m}Xe	15.6 min	2.28E+05	8.3E-06	0.8	0.8
^{89}Rb	15.4 min	2.88E+04	6.8E-06	0.7	0.6
^{138}Cs	32.2 min	2.59E+04	6.4E-06	0.6	0.6
^{91}Y	58.8 d	1.71E+02	5.9E-06	0.6	0.6
^{137}Cs	30.1 y	8.81E+02	5.2E-06	0.5	0.5
^{239,240}Pu	24390 y	1.07E-01	5.1E-06	0.5	0.5
^{133}I	20.9 h	1.69E+03	4.5E-06	0.4	0.4

[a] Of the radionuclides evaluated, these radionuclides contributed more than 99% to the total screening value; the highlighted radionuclides represent more than 95% of the screening value for onsite exposure at Highway 20 from the inhalation and plume immersion pathways.

Establishing Relative Importance of Years

For offsite exposures, the screening assessment showed the individual years 1957, 1958, and 1959 had the highest screening values (Figure 17). At the onsite location, 1963, 1964, and 1965 had the largest screening values (Figure 18). These same years, 1963–1965, had the highest ^{41}Ar and ^{138}Xe releases and exposure to these radionuclides was by plume immersion, one of the two main pathways of exposure at the onsite location.

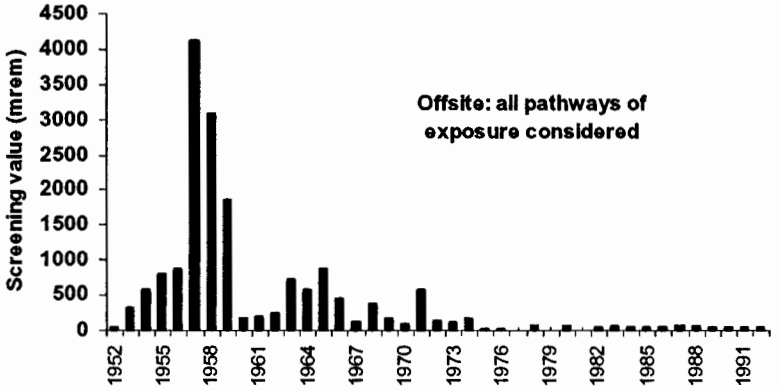

Figure 17. Results of screening individual years for all radionuclides for all pathways; results from 1952–1992 are shown. The screening value provides a relative way to evaluate the years that had the largest contribution to offsite screening value. The years 1957, 1958, and 1959 had the highest airborne releases contributing to the screening value. To convert the screening value to Sv, divide the screening values by 100,000.

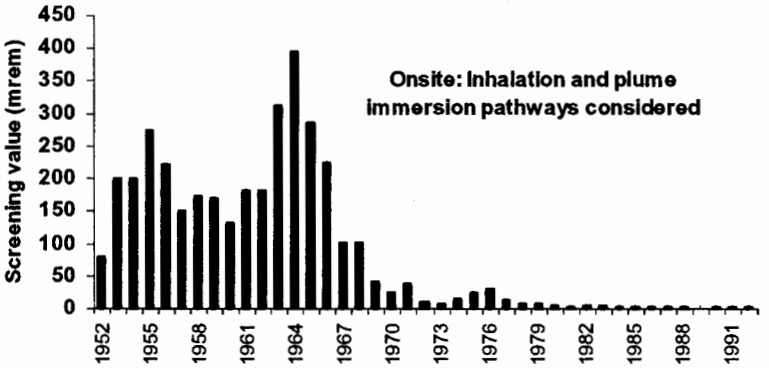

Figure 18. Results of screening individual years at the onsite location from all radionuclides for exposure by the inhalation and plume immersion pathways. The screening method provided a relative way to evaluate the years that contributed the most to the screening values from routine releases of radionuclides. The years 1963, 1964, and 1965 are the years with highest airborne releases contributing to the screening value at the onsite location. To convert to Sv, divide the screening values by 100,000.

When we looked at the screening results for the important individual years, the radionuclides that emerged as the most main contributors were similar to those radionuclides that emerged when all years were screened (see Tables 6 and 7). However, releases during the RaLa processes, which occurred during the high release years of 1957, 1958 and 1959, shifted the focus of the most important radionuclide from ^{137}Cs to ^{131}I for offsite exposure (Table 8a). The effect of the releases from the RaLa runs was seen onsite as well, with ^{131}I and ^{90}Sr the chief contributors to the total screening value in 1957 (Table 8b). In contrast, ^{41}Ar, and several xenon and krypton

isotopes were the top contributors when all years were considered (see Table 7). Nevertheless, ^{131}I and ^{90}Sr were still major contributors to the screening value when all years were evaluated. In summary, of the radionuclides released to air from the INEEL in 1957, ^{131}I, ^{137}Cs, and ^{90}Sr were the main contributors to the screening value for offsite exposures at Atomic City. At the onsite exposure location, ^{41}Ar, several krypton and xenon isotopes, ^{144}Ce, ^{90}Sr, ^{106}Ru, and ^{131}I were the main contributors to the screening value for the inhalation and plume immersion pathways.

Table 8a. Radionuclides Contributing Over 95% to the Total Screening Value in 1957 at the *Offsite Location* [a]

Radionuclide	1957 releases (Ci)	Screening value (Sv)	Screening value (mrem)	Percent of total screening value
^{131}I	1400	0.029	2900	70
^{137}Cs	36	0.0059	590	15
^{90}Sr	36	0.0051	510	12

[a] For the offsite exposure to routine releases, 1957 had the highest screening value.

Table 8b. Radionuclides Contributing 95% to the Total Screening Value in 1957 at the *Onsite Location* [a]

Radionuclide	1957 releases (Ci)	Screening value (Sv)	Screening value (mrem)	Per cent of total screening value
^{131}I	1400	0.00058	58	31
^{90}Sr	36	0.00035	35	19
^{41}Ar	34,000	0.00018	18	10
^{144}Ce	54	0.00015	15	8
^{238}Pu	0.078	0.00013	13	7
^{138}Xe	18,000	0.000089	8.9	5
^{133}Xe	460,000	0.000069	6.9	4
^{88}Kr	5500	0.000055	5.5	3
^{132}I	4000	0.000054	5.4	2.9
^{147}Pm	140	0.000040	4.0	2.1
^{133}I	440	0.000032	3.2	1.7
$^{239,240}Pu$	0.012	0.000022	2.2	1.2

[a] For the offsite exposure to routine releases, 1957 had the highest screening value of all years.

Tables 9a and 9b show results for 1964, the year with the highest screening value at the offsite and onsite locations, respectively. However, in 1964, ^{106}Ru emerged as the top contributor to the screening value along with ^{90}Sr and ^{137}Cs. Again these results can be compared to the results obtained when all years were screened (Tables 6 and 7). High releases of ruthenium from the WCF at the ICPP occurred in 1964 and these releases were evaluated as an episodic release.

Table 9a. Radionuclides Contributing 95% to the Total Screening Value in 1964 at the Offsite Location [a]

Radionuclide	1964 releases (Ci)	Screening value (Sv)	Screening value (mrem)	Percent of total screening value
^{106}Ru	340	0.0025	250	42.2
^{90}Sr	8.8	0.0012	120	21.3
^{137}Cs	4.8	0.00079	79	13.5
^{41}Ar	260,000	0.00029	29	4.9
^{129}I	0.55	0.00023	23	4.0
^{138}Xe	140,000	0.00022	22	3.8
^{14}C	610	0.00012	12	2.0
^{60}Co	0.72	0.000091	9.1	1.6
^{134}Cs	0.93	0.000090	9.0	1.5

[a] For the onsite exposure to routine releases, 1964 had the highest screening value.

Table 9b. Radionuclides Contributing 95% to the Total Screening Value in 1964 at the Onsite Location [a]

Radionuclide	1964 releases (Ci)	Screening value (Sv)	Screening value (mrem)	Percent of total screening value
^{41}Ar	260,000	0.0013	130	34
^{106}Ru	340	0.0012	120	30
^{138}Xe	140,000	0.00066	66	17
^{88}Kr	41,000	0.00041	41	10
^{87}Kr	42,000	0.00014	14	3.6
^{144}Ce	23	0.000063	6.3	1.6

[a] For the onsite exposure to routine releases, 1964 had the highest screening value.

Establishing Relative Importance of Facilities

Finally, we evaluated the main facilities at the INEEL for their relative annual contribution to the offsite screening value. We calculated the total screening value for each facility by year by summing the screening values for individual radionuclides for that year. Figure 19 shows results for screening selected years for three main facilities at the INEEL: ICPP, TRA, and ANL-W. The screening results show clearly that the ICPP dominated the releases of radionuclides that contributed to the screening value, and was important in terms of human health as ascertained by the NCRP screening methods. Although TRA released the highest levels of radioactivity from the INEEL in the early 1950s and mid1960s (see Figure 3), the radionuclides released from the ICPP during those times were more important in terms of potential offsite doses, as measured by the screening values.

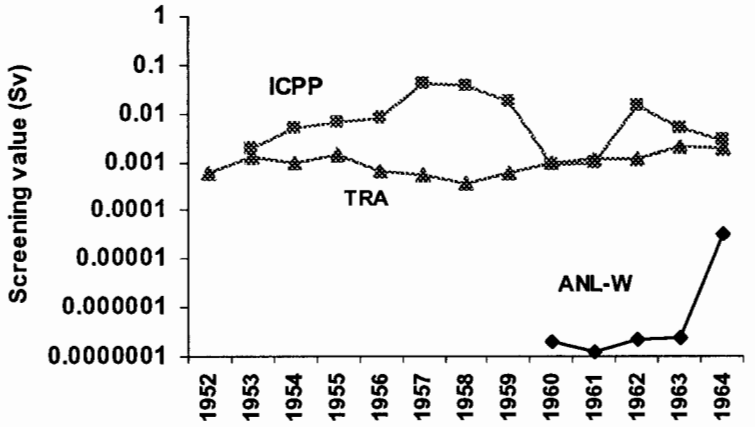

Figure 19. Comparison of total screening values for radionuclides released from ICPP, TRA, and ANL-W from 1952 through 1964. The consistently high screening values for ICPP demonstrated its important and continuing role in contributing to the screening value.

Table 10 summarizes information about the main source of the key radionuclides and the important pathways of exposure for each. Clearly, the ICPP has been the most important source historically for these key radionuclides released to air from routine releases. Of the radionuclides that contributed greater than 95% of the screening value from our screening assessment, ^{131}I, ^{137}Cs, ^{90}Sr, and ^{41}Ar were important in all.

Table 10. INEEL Source Of Primary Radionuclides Of Concern

Radionuclide	Main historical INEEL source	Main pathways of exposure for radionuclide
^{144}Ce	ICPP	• Ingestion of produce • Inhalation
^{131}I	ICPP	• Ingestion of milk, meat, vegetables
^{106}Ru	ICPP	• Ingestion of produce • Ground contamination
^{90}Sr	ICPP	• Ingestion of milk, meat, vegetables
^{137}Cs	ICPP	• Ground contamination • Ingestion of meat, milk, produce
^{134}Cs	ICPP	• Ground contamination • Ingestion of meat, milk, produce
^{238}Pu, $^{239,240}Pu$	ICPP	• Inhalation • Ingestion of vegetable
^{41}Ar	TRA	• Plume immersion

GROUNDWATER PATHWAY

The 890 square mile area that makes up the INEEL overlies the Snake River Plain Aquifer, which is the primary source of drinking water for most of eastern Idaho. Starting in 1953, wastewater containing radioactive and chemical contaminants was released into the aquifer through both injection wells and disposal ponds. Liquid radioactive waste was disposed of using these methods at the ICPP, TRA, and TAN. At ICPP, a 600-foot deep injection well was used to dispose of radioactive waste from 1953 to 1984. In 1984, the well was closed and replaced by waste seepage ponds. At TRA, radioactive waste was disposed of using waste ponds starting in 1952. From 1953 to 1972, liquid radioactive waste was discharged from TAN to the aquifer through a 310-foot injection well. An infiltration pond replaced the well in 1972. Additionally, some of the contaminants in waste buried at the RWMC have migrated into the groundwater.

A monitoring system of wells for the Snake River Plain Aquifer has been operated by the USGS since 1949. The original purpose of the monitoring wells was to assess the water resources of the area before constructing facilities at the INEEL, but the USGS has maintained the network of samplers to determine hydrologic trends and assess the movement of facility-related contaminants in the aquifer (Bartholomay et al. 1995). A report series has been produced by the USGS to document hydrologic conditions every few years. This set of reports provides the most comprehensive series of groundwater monitoring data for the aquifer and was used to complete these screening calculations for the groundwater pathway at the INEEL. Site environmental reports also document groundwater contamination, but they focus mostly on offsite contamination and refer to the USGS reports for onsite contaminants in groundwater. Radioactive contaminants that have been detected in the groundwater include ^{3}H, ^{90}Sr, ^{60}Co), ^{137}Cs, ^{129}I, ^{238}Pu, $^{239,240}Pu$, and ^{241}Am. Analyses were also done for chromium-51, but this radioactive contaminant has never been detected in the aquifer.

Hydrology and Geology of the Site

It is estimated that the Snake River Plain Aquifer may contain more than 1 billion acre-feet of water (Barraclough et al. 1982). Movement of groundwater in the aquifer is generally from northeast to southwest, eventually discharging to springs along the Snake River 100 miles southwest of the INEEL. The velocity of the water ranges from 5 to 20 feet per day.

The aquifer is made up of fractured basaltic lava flows and interbedded sedimentary deposits. The water in the aquifer is contained in intercrystalline and intergranular pores, cavities, fractures, etc. (Pittman et al. 1988).

It is this hydrogeology that has caused a number of perched groundwater zones to form at the INEEL. A perched groundwater zone forms when downward flow to the aquifer is impeded by silt and clay in the sedimentary units or by dense basaltic flows (Pittman et al. 1988). Perched groundwater zones have formed in areas where liquid waste is disposed of using infiltration ponds. Water from these ponds percolates into the alluvium and is perched by fine-grained sediment near the base of the alluvium, approximately 50 feet below the land surface. These perched groundwater zones are typically about twice the size of the ponds under which they lie.

The water perched in these zones further percolates into the basaltic rocks until it reaches the bottom of a sedimentary deposit, which extends from about 100 to 150 feet below land surface. The water is then transmitted through the unsaturated basalt to the aquifer by the unsaturated

basaltic rocks and other sedimentary deposits that underlie these zones. The perched water zones are then recharged by water from the disposal ponds, irrigation water, and infiltration of natural precipitation. The sedimentary interbeds slow the migration of water from these zones into the aquifer and contaminant concentrations are diluted by water from other sources. Water travel time is generally controlled by the presence of the interbeds (Magnuson and Sanders 1998) or dense basalt. Flow through the fractured basalt is relatively rapid. Water travel times to the aquifer under the RWMC have been estimated to range from 20 to 90 years (Magnuson and Sanders 1998).

Contaminants Measured in the Snake River Plain Aquifer

Analysis of groundwater for more than 20 chemicals and radionuclides has resulted in the detection of a number of radioactive contaminants, including tritium, ^{90}Sr, ^{60}Co, ^{137}Cs, ^{129}I, ^{238}Pu, $^{239,240}Pu$, and ^{241}Am. These contaminants have been carefully studied and measured by the USGS, and those data have been used in this analysis.

Plutonium Isotopes

Monitoring of plutonium isotopes discharged to the Snake River Plain Aquifer began in 1974. Before that time, monitoring techniques were not capable of distinguishing plutonium isotopes from gross alpha radiation. As early as 1975, concentrations of plutonium were detectable in regional groundwater at a well in the immediate vicinity of the ICPP disposal well. The concentrations were several orders of magnitude lower than the concentrations measured in the waste discharged to the well. It was postulated at that time that the concentration reduction over a short distance in the aquifer was due to dilution, dispersion, and removal of the soluble nuclides by sorption (Barraclough et al. 1982).

Plutonium measured in subsequent samples from this well and three others in the same area tended to support the same conclusions (Pittman et al. 1988; Orr and Cecil 1991; Bartholomay et al. 1995). Through January 1987, some samples from these wells still showed plutonium above the reporting level, but since then, no well has shown any detectable concentration of plutonium (Bartholomay et al. 1995). The disposal well at TAN showed some detectable concentrations of plutonium in the late-1980s, but again, the concentrations were low and no spread of the contamination has been detected in the aquifer. The plutonium is highly sorbed into sediments and does not remain in solution. Because plutonium has not been detected in any groundwater wells outside of the immediate vicinity of disposal areas, we have determined that the exposure pathway to plutonium via groundwater was not a complete offsite exposure pathway for this historical screening assessment.

Americium-241

Americium-241 is a radioactive decay product of ^{241}Pu. Both wastewater discharged to the aquifer and wastes buried at the RWMC have contained plutonium isotopes and, consequently,

also ^{241}Am. Concentrations of ^{241}Am exceeding the reporting level[8] were detected at four wells in the area of the RWMC and in the TAN disposal well between 1972 and 1988. No measurable concentrations have been detected offsite or outside the regions of RWMC and TAN. Since 1988, no detectable concentrations of ^{241}Am have been measured in any wells in the aquifer at the INEEL. Based on this analysis, the exposure pathway to ^{241}Am via groundwater was not a complete offsite exposure pathway for this historical screening assessment.

Cesium-137

Cesium-137 has been disposed of in INEEL wells and disposal ponds since 1952, but before 1982, cesium was not detected in any of the observation wells. In 1982 and 1983, ^{137}Cs was detected in six water samples collected from the well nearest the ICPP disposal well. No future samples contained detectable concentrations of ^{137}Cs. In 1984 and 1985, two water samples from a nearby well contained ^{137}Cs, but samples from those wells have had no detectable cesium since that time.

The absence of detectable ^{137}Cs concentrations has been attributed to the discontinuation of the use of the ICPP disposal well, as well as removal of cesium from solution by sorption to the alluvium, sedimentary interbeds, and basalt (Pittman et al. 1988; Bartholomay et al. 1995). We determined that this pathway was not a complete offsite exposure pathway for this historical screening assessment.

Cobalt-60

Concentrations of ^{60}Co exceeded the reporting level at only one observation well south of TAN throughout the history of the site. This well contained detectable amounts of ^{60}Co in some samples during 1982-1985. Since 1985, no ^{60}Co has been detected in any Snake River Plain Aquifer observation well. Contributing to the lack of ^{60}Co in aquifer water samples are reduction in discharge, radioactive decay, and sorption processes in the unsaturated perched water ground. The groundwater pathway for ^{60}Co was not a complete offsite exposure pathway for this historical screening assessment.

Iodine-129

Iodine-129 was disposed of through the ICPP disposal well from startup in 1952 through closure of the disposal well in 1984 (Barraclough et al. 1982; Lewis and Jansen 1984). In 1977, concentrations of ^{129}I were measured in the aquifer onsite at levels ranging from 0.9 to 27 picocuries per liter (pCi L^{-1}). The concentrations were highest near the ICPP disposal well. The ^{129}I plume in groundwater had migrated about 3 miles from the ICPP disposal well in 1977. By 1981, further migration of the ^{129}I plume had been noted, and the plume was about 6.3 miles from the ICPP disposal well (about 2 miles from the site boundary), with concentrations ranging from 0.05 to 41 pCi L^{-1}. This plume migration was quite surprising, because of the small amount of

[8] The reporting level is defined as a concentration in aquifer water that exceeds the sample standard deviation by 3 times. Concentrations below this level were considered to be below the minimum detection limit, which means that contamination was not detected at a statistically significant level.

iodine disposal from 1977–1981. During the years between 1977 and 1981, iodine detection techniques improved considerably, and it was postulated that the improvement in capabilities made detection of smaller quantities possible and, therefore, the edges of the plume small were easier to detect.

To prove this hypothesis, the scientists looked at the concentrations in comparable parts of the plume from 1977 to 1981, and they discovered that the concentrations did not vary much at all. The only exceptions to this were at the wells closest to the disposal well, where lower concentrations were seen in the 1981 plume than in the 1977 plume. This observation seemed to support the small releases and the hypothesis of no plume spread but rather improved plume detection (Lewis and Jansen 1984).

Groundwater samples collected in August 1986 showed decreased concentrations of ^{129}I in onsite wells, which would be expected because the disposal well was no longer in use. Additionally, the plume seemed to have "receded" by 1986 to within 5.6 miles of the Site boundary (Chew and Mitchell 1988). This was probably not a plume recession, but rather a reduction in concentration such that the plume edges could no longer be detected.

From the available information, we believed that the ^{129}I plume has not gone offsite, and therefore did not create a complete exposure pathway for the offsite individual. Because of the long half-life of the radionuclide (~16.4 million years), however, the plume will probably be present in some form onsite for an extended period of time.

Strontium-90

Strontium-90 was discharged to the ICPP disposal well from 1952 to 1984 and to the infiltration ponds at the ICPP after the disposal well was closed. The ^{90}Sr plume in the groundwater has been measured since the early 1970s. In 1978, the plume covered about 2.2 square miles and was detected less than 3 miles southwest of the ICPP disposal well, with concentrations in the aquifer ranging from 24 to 93 pCi L^{-1}, and with higher concentrations occurring closer to the well. By 1985, the plume size had not changed appreciably, but the concentration in the plume had decreased by about 10 pCi L^{-1} since 1981 near and south of ICPP because of discontinued used of the disposal well. Aquifer concentrations in 1985 ranged from 6 to 63 pCi L^{-1}. The plume size had decreased to about 0.8 square miles by 1988, with concentrations decreasing another 33 pCi L^{-1}. There was no appreciable change in concentration or size of the plume between 1988 and 1991, primarily because of lack of recharge of the aquifer from the Big Lost River.

Because the plume never approached the INEEL boundary and has diminished in size and concentration over the years, the groundwater pathway for ^{90}Sr was not considered to be a complete offsite exposure pathway for this screening analysis.

Tritium

The disposal of tritium in liquid effluents has been monitored at the INEEL since 1961. Much of the tritium was discharged directly into the aquifer through the ICPP disposal well, while other quantities were discharged to disposal pits and percolated down into the groundwater. Because tritium in solution forms tritiated water, an analog to water, it moves easily through water systems.

The disposal of tritium has resulted in a large, dispersed plume in the Snake River Plain Aquifer. In 1978, the plume was estimated to cover about 28 square miles, with the highest tritium values occurring around the ICPP disposal well and decreasing at greater distances from the well. The plume at that time had migrated about 7.5 miles downgradient from the well at an average rate of 4 to 5 feet per day (Barraclough et al. 1982). By 1981, the plume size had increased to about 42 square miles and by 1985 to about 51 square miles. Because of the discontinued use of the ICPP disposal well, however, aquifer concentrations in 1985 ranged from 0.9 to 93 pCi mL^{-1}, a decrease from 1981 concentrations of 0.4 to 156 pCi mL^{-1}.

During 1983-1985, tritium was detected near the southern boundary of the INEEL in the groundwater supply for the first time. Three wells (Figure 20) located along the southern boundary of the INEEL had detectable concentrations of tritium: wells 103, 105, and 108. Well 103 had a tritium concentration of 0.8 pCi mL^{-1} in July 1983 and 1.2 pCi mL^{-1} in July 1985. In January 1984, tritium was detected at a concentration of 0.5 pCi mL^{-1} in well 105. In October 1985, well 108 showed a tritium concentration of 0.8 pCi mL^{-1}. No further quarterly samples at these or any other boundary locations verified the presence of tritium. The maximum concentration level for tritium in drinking water is 20 pCi mL^{-1}.

By 1988, the tritium plume in groundwater had decreased in size from 51 to 45 square miles and concentrations were reduced to about 0.7 to 61.6 pCi mL^{-1}. Further reductions in both plume size and plume concentration were noted by 1991. Concentrations decreased by as much as 23 pCi mL^{-1}, and the size of the plume was further reduced to 40 square miles (Orr and Cecil 1991; Bartholomay et al. 1995).

After the 1983-1985 detection of tritium in observation wells at the southern boundary of the INEEL, no more tritium was detected offsite in groundwater. It has been shown that the tritium plume has receded, and it is speculated that radioactive decay (tritium has a half-life of about 12.3 years), reduction in tritium disposal rates, dilution from recharge of the aquifer, and changes in disposal methods have contributed to the plume recession and reduction in total concentration.

Because tritium in the groundwater was detected at the site boundary during the years 1983, 1984, and 1985, it was important to complete screening calculations for this pathway to determine if the dose and risk associated with it warranted further investigation.

Groundwater Screening Calculations

Tritium concentrations in the groundwater were detected at the site boundary of the INEEL at different times during 1983-1985. We developed a scenario here to assess the dose and risk associated with potential exposure to tritium in offsite groundwater.

For screening purposes, we used two separate screening models. The NCRP screening models implement a dose-based screening methodology (NCRP 1996). A version of this model contains a component for determining dose from surface water ingestion, which was adapted here for use as a groundwater intake model, assuming that the concentration in the groundwater was ingested. The screening factors used in this model were developed based on screening level intakes and established dose conversion factors. The dose calculated as an endpoint to this model was available for comparison to other pathway doses produced during this screening analysis or to annual dose limits.

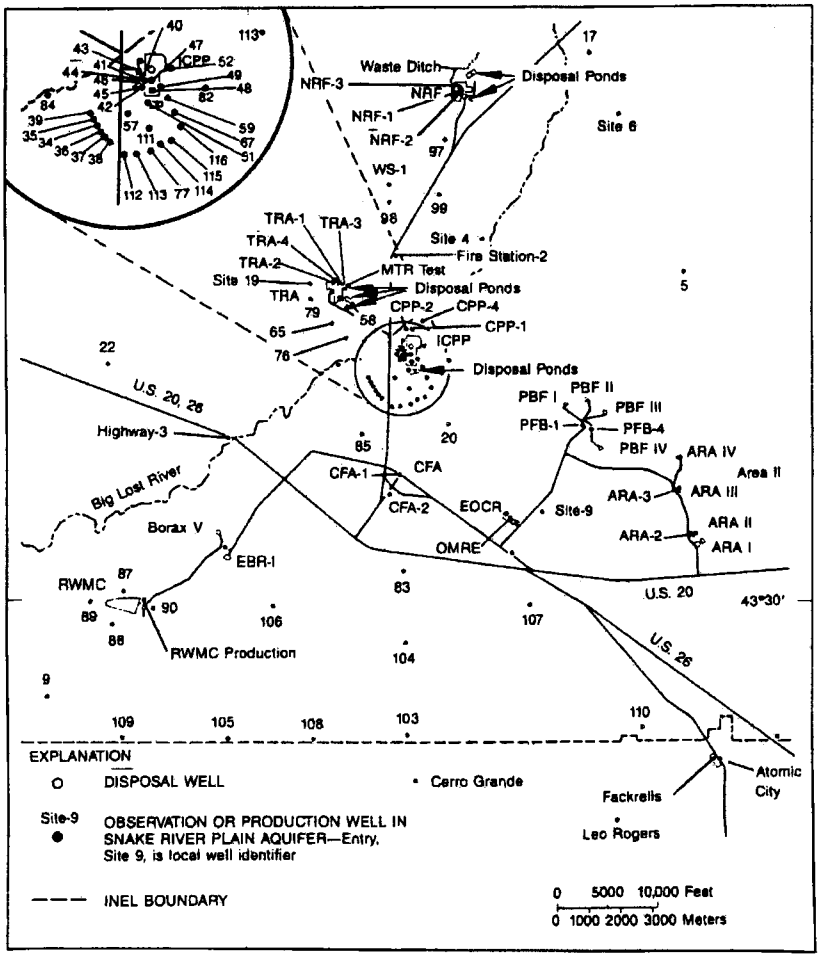

Figure 20. Snake River Plain Aquifer observation wells at the INEEL. Wells 103, 105, and 108, closest to the INEEL southern boundary (dotted line), are the wells that contained measurable quantities of tritium in 1983-1985.

A second screening strategy used here involved the EPA risk-based screening models (EPA 1999). These models provide intake-to-risk conversion factors for cancer mortality and incidence. Screening level intakes as suggested by the NCRP were used in task Order 5 to assess risks produced by this model. Because this is a tap-water intake model where the user provides the tap water concentration data, we assumed that the well concentrations reflect concentrations at the tap. The risks produced as endpoints to this model could then be compared to risk levels proposed by EPA and other agencies as appropriate screening levels for risk.

Concentrations in the boundary wells ranged from 0.5 to 1.2 pCi mL^{-1} during the 1983-1985 period. To make the calculation conservative, we assumed that the concentration in groundwater

at the southern boundary of the INEEL was 1.2 pCi mL^{-1} for the 5-year period from 1981-1985[9]. The concentration in groundwater was important for both calculations, while the span of time over which that concentration existed was important only for the risk calculation. Dose was calculated on an annual basis, but risk was calculated over a lifetime of exposure, so it was important to know the duration of exposure.

Screening Dose Calculation

A screening dose is generally calculated using source term estimates and calculating a downstream concentration, but because we had measurements of tritium in groundwater, we used these values as our concentration estimates. The assumed concentration of tritium in water was 1.2 pCi mL^{-1}. To convert this value to becquerels per cubic meter (Bq m^{-3}), we multiplied by 37,000, so the equivalent concentration was 44,000 Bq m^{-3}. The screening factor for tritium, developed by NCRP based on annual intake and dose conversion factors for tritium ingestion, is 1.4×10^{-11} sievert (Sv) per Bq m^{-3} (NCRP 1996). Multiplying these two values gives 6.2×10^{-7} Sv. This is the dose for 1 year, and is equivalent to 0.06 mrem.

$$1.2 \text{ pCi mL}^{-1} \cdot 37000 \frac{\text{Bq m}^{-3}}{\text{pCi mL}^{-1}} = 44000 \text{ Bq m}^{-3} \cdot 1.4 \times 10^{-11} \text{ Sv per Bq m}^{-3}$$

$$= 6.2 \times 10^{-7} \text{ Sv} = 0.06 \text{ mrem}$$

For perspective on the magnitude of this dose, we compared this to the annual dose limit for drinking water exposure, which is 10 mrem. The annual dose calculated here assumes a very conservative water ingestion rate of 800 liter per year (L y^{-1}) as well as a conservative concentration of tritium in water because the 1.2 pCi mL^{-1} value was only measured in one quarterly sample. Still, the dose calculated here is significantly less than the annual dose limit of 10 mrem. This dose can be compared to other screening doses calculated in this report.

Screening Risk Calculation for Groundwater Exposure

Risk-based screening calculates lifetime risk of cancer mortality or incidence from ingestion of radioactivity. Detectable concentrations were only measured offsite during different months over the course of 3 years, but we assumed the maximum offsite concentrations existed for 5 years to be conservative. We also employed the conservative ingestion rate of 800 L y^{-1} used in the NCRP calculations. The assumed concentration of tritium in water was 1.2 pCi mL^{-1} (equivalent to 44.4 [becquerels per liter] Bq L^{-1}). The total intake of water during the 5-year exposure period would be 4000 L. Multiplying the product of these two values by mortality and incidence risk coefficients of 9.44×10^{-13} Bq^{-1} and 1.37×10^{-12} Bq^{-1}, respectively, gave a mortality lifetime risk of 1.7×10^{-7} and a incidence lifetime risk of 2.4×10^{-7}.

[9] It is important to note that the groundwater concentration used for this screening calculation was only measured offsite at one well during one month in 1985, as described in the preceding text. Two other wells had concentrations less than this level during different months, but no well exhibited continuously elevated concentrations of tritium. We are using this concentration for such an extended period of time to conduct a screening calculation, not to make a realistic assessment of dose.

$$1.2 \text{ pCi mL}^{-1} \cdot 0.037 \frac{\text{Bq}}{\text{pCi}} = 0.044 \text{ Bq mL}^{-1} \cdot 1000 \frac{\text{mL}}{\text{L}} \cdot 800 \text{ L y}^{-1} \cdot 5 \text{ y} = 1.78 \times 10^{5} \text{ Bq}$$

$$1.78 \times 10^{5} \text{ Bq} \cdot 9.44 \times 10^{-13} \text{ Bq}^{-1} = 1.7 \times 10^{-7} \text{ lifetime mortality risk}$$

$$1.78 \times 10^{5} \text{ Bq} \cdot 1.37 \times 10^{-12} \text{ Bq}^{-1} = 2.4 \times 10^{-7} \text{ lifetime incidence risk}$$

For perspective on these risk values, following is some information on comparative screening factors. During the Oak Ridge Environmental Dose Reconstruction, an increased lifetime cancer incidence risk criterion of 10^{-5} was applied for screening releases of radionuclides to the aquatic pathways (Apostoaei et al. 1999). In the Hanford Environmental Dose Reconstruction project, one of the criteria used to define the physical area to be included in the study calculations (study domain) was a thyroid dose of 1 rad (0.01 Gy) to a child or infant (Shleien 1992). This dose represents an increased lifetime risk for radiation induced thyroid cancer on the order of 2×10^{-4}. The EPA has specified an upper bound individual lifetime cancer risk "target range" for carcinogens of 10^{-4} to 10^{-6} within which they strive to manage risks as a part of a Superfund cleanup. The risk estimates are determined using reasonable maximum exposure assumptions for either current or future land use (EPA 1991). The EPA approach was adapted to identify and prioritize potential remediation sites at the INEEL using a target risk level of 10^{-6} (Fromm 1996).

These other studies and agencies set the risk screening criteria somewhere in the range 10^{-4} to 10^{-6} for remediation and dose reconstruction efforts, and may be a guide for other sites. This means that risks higher than this range would require further investigation, and risks lower than this range would be minimal enough to be eliminated from consideration for further study. The risk calculated here for offsite tritium exposures to groundwater at the INEEL was nearly an order of magnitude lower than the lowest limit of this proposed range. Again, it is important to stress that this risk was calculated conservatively, assuming that the highest concentration ever measured offsite was present continuously for 5 years. The data do not indicate that a concentration of this magnitude was present for such an extended period of time. This analysis only indicates dose and risk for offsite exposures to groundwater in the past, and does not make any judgments regarding onsite exposures or future offsite exposures to other nuclides.

EPISODIC RELEASES

A number of tests, accidents, and other events at the INEEL have led to episodic or short-term releases of radionuclides to the atmosphere. The Task Order 5 work evaluated these episodic releases, and ranked them according to the potential for exposure to members of the public, and also determined the specific radionuclides that are the most important contributors to potential exposure. This effort will help focus future work on those events with the greatest probability of delivering the largest dose to a member of the public. Potential for exposure was evaluated by applying atmospheric screening factors (NCRP 1996) to the amount or quantity of individual radionuclides released for a given episodic release.

Selection Of Important Radionuclides

DOE (1991a) evaluated episodic releases from INEEL facilities and used screening calculations to narrow the list of radionuclides to those that were the most important in terms of potential dose to exposed members of the public. Fission product inventories were calculated for both short, transient nuclear power operations and sustained, long-term nuclear power operations. The inventories were calculated using RSAC-3 (Wenzel 1982), a computer code designed to calculate fission product inventories based on various reactor operational scenarios. The transient operation assumed by DOE (1991a) lasted for 1.5 msec at a power level of 30,000 MW. The transit time to the INEEL Site boundary for the released radionuclides was assumed to be 2.9 hours (174 minutes). The sustained operation was assumed to last for 32.1 days at a power level of 1.48 MW, followed by a 40-day decay time for the created fission products. Radionuclides with zero inventory following the first decay time (2.9 hours for the transient operation calculation and 40 days for the sustained operation calculation) and radionuclides with half-lives less than 10 minutes were deleted from the list. The remaining radionuclides were then sorted by their relative contribution to both inhalation and immersion dose, based on ICRP dose conversion factors. The most important contributors to dose were selected, resulting in a list of 47 radionuclides (DOE 1991a). The activation product, ^{41}Ar, and fuel element constituents, ^{234}U, ^{235}U, and ^{238}U, were also added to this list. This was done to account for the potential activation of naturally occurring stable argon by neutrons from the reactor and the potential release of some fraction of the actual fuel element(s).

For the Task Order 5 screening analysis, we used a slightly different approach because some of the episodic releases had transit times to the Site boundary as short as 0.3 hours (DOE 1991a). We decided that the selection of radionuclides present following a 2.9 hour decay time and the deletion of radionuclides with half-lives of less than 10 minutes was inappropriate. We developed a more inclusive list of radionuclides using a slightly different approach that allowed for evaluating the potential importance of the shorter-lived radionuclides to which members of the public may have been exposed.

Using the RSAC-5 computer code (Wenzel 1994), we calculated fission product inventories based on the transient and sustained operational scenarios described above. The RSAC-5 computer code calculates quantities of direct fission products as well as daughter products arising from the subsequent decay of fission products. Tables 11 and 12 show these two assumed operational scenarios along with, for comparison to the assumed scenarios, reactor operating parameters for actual transient operations and accidents (Table 11) and sustained operations (Table 12) that have resulted in atmospheric releases at the INEEL. The transient operation selected for this screening analysis was based on the SNAP 10A Transient (SNAPTRAN)-3 reactor operation and was similar to other transient operations that have resulted in episodic releases at the INEEL. The sustained operation selected for this screening analysis is based on typical MTR operations and was similar to sustained reactor operations involved in episodic releases at the INEEL. Our analysis assumed that the operational scenarios used for the radionuclide selection calculations adequately represented actual operations at the INEEL. To test the validity of this assumption, we evaluated fission product inventories using different power levels and operating times. Changing these parameters resulted in slightly altering the ranking order of the most important radionuclides in some cases, but it did not cause the appearance of

any additional radionuclides that were not selected based on the operational scenarios that we assumed for this ranking and selection process.

Transient Operation

For the transient operation, we calculated inventories at 20, 60, 120, 300, and 600-minute decay times to allow for ingrowth of decay or daughter products. We assumed release fractions of 0.1 for solids, 0.5 for halogens, and 1.0 for noble gases and deleted all radionuclides with zero inventory at a decay time of 20 minutes. This resulted in a list of 233 radionuclides.

Table 11. Transient Reactor Operations and Accidents at the INEEL Resulting in Episodic Atmospheric Releases

Assumed operation	Power (MW)	Operating time (s)	Burn-up (MW-s)
Assumed transient operation	30,000	0.0015	45
Comparable actual operations			
ICPP 1959 criticality	1,282[a]	1.0	1282
ICPP 1961 criticality	20[a]	1.0	20
SNAPTRAN-3	30,000	0.0015	45
SNAPTRAN-2	36,000	0.0015	54
SPERT-I, #1	10,000	0.0032	32
SPERT-I, #2	70,500	0.0022	155
SPERT-I, #3	106,500	0.00155	165
BORAX-I	52,000	0.0026	135

[a] Power level based on estimated number of fissions during criticality

Table 12. Sustained Reactor Operations at the INEEL Resulting in Episodic Atmospheric Releases

Assumed operation	Power (MW)	Operating time (d)	Burn-up (MW-d)	Decay (d)
Assumed sustained operation[a]	1.5	32	48	40
Comparable actual operations				
FEBT-A	0.56	5	2.8	70
FEBT-B	0.56	69	39	250
FPFRT-1	0.07	19	1.4	922
FPFRT-2	0.066	19	1.3	934
FPFRT-3	0.07	19	1.4	932
FPFRT-4	0.061	19	1.2	942
FPFRT-5	0.034	5	0.18	43
FPFRT-6	0.034	5	0.18	51
FPFRT-7	0.034	5	0.18	64
FPFRT-8	0.034	5	0.18	65
FPFRT-9	0.077	69	1.5	985

[a] Based on typical MTR operations

To focus on the potentially most important radionuclides, we evaluated the radionuclides present at the above decay times using atmospheric screening factors provided by NCRP (1996). The RSAC-5 computer code calculates inventories of fission radionuclide decay products, or daughters, and the NCRP screening factors include daughter product contribution assuming a 30-year buildup period in the environment. Because the RSAC-5 computer code calculates inventories of fission product daughters and because it was not appropriate to assume a 30-year buildup period for episodic releases, we used the screening factors for the parent radionuclide only, excluding the contribution from daughter products given by NCRP (1996). Multiplying the inventory or quantity of a given radionuclide by the appropriate screening factor enabled us to calculate a value that was used to rank the list of radionuclides according to their potential contribution to radiation dose to an exposed individual at a given decay time. The screening values were summed for all radionuclides present at a given decay time, and each radionuclide was assigned a relative dose percentage. The list was then sorted from highest to lowest percentage for each decay time.

It is important to note that the NCRP screening factors were "...designed to be used for intermittent or continuous routine long-term releases from point sources only that are assumed to occur over a period of a year" (NCRP 1996). While the screening calculation described above does not result in an estimated dose, the screening factors can be used to select the most important radionuclides because the screening factors are proportional to the effective dose factors from which they were derived.

Using the NCRP atmospheric screening factors in this way, though, may introduce certain biases into the selection process. For example, the screening factors assume an average annual air concentration and a 30-year buildup time, which was not the case for the episodic releases. This means that the contribution via the ingestion and ground irradiation pathways in particular may be overestimated somewhat because they are the two pathways most impacted by an extended period of buildup. This is primarily true for the longer-lived radionuclides (e.g., ^{137}Cs and ^{90}Sr), but the potential importance of shorter-lived radionuclides may also be exaggerated to some extent because the screening factors assume an annual average air concentration, which was not the case for the episodic releases. However, we do not believe these limitations preclude our use of the screening factors for the screening calculations, and we maintain that they represent the most logical and efficient approach for focusing our efforts on the most important releases and radionuclides. We examine these potential biases further in a later section of this report.

We elected to use the *total*, *inhalation*, and *plume immersion* screening factors reported by NCRP (1996) for our selection methodology because these are the three pathways of importance for evaluating the actual releases. The *total* screening factors are most appropriate to use for evaluating offsite exposure at a potential residence location because they include contributions from all pathways. The *inhalation* and *plume immersion* screening factors are most appropriate to use for evaluating shorter duration onsite exposures, such as might have occurred to a utility worker or motorist along a publicly accessible roadway passing through the Site.

Several of the short-lived radionuclides (generally radionuclides with half-lives less than 10 minutes) do not have corresponding screening factors. To evaluate the potential importance of these short-lived radionuclides, we selected a conservative **surrogate** screening factor to apply to these radionuclides in the calculated fission product inventory. The largest **existing** *total* screening factor for radionuclides with half-lives of less than 1 hour (^{130}Sb) was selected and applied to all radionuclides without existing screening factors. Similarly, the largest **existing**

inhalation and *plume immersion* screening factors for radionuclides with half-lives less than 1 hour (^{133m}Te for inhalation and ^{130}Sb for immersion) were selected and applied to all radionuclides without existing screening factors.

For each decay time, those radionuclides with existing screening factors and contributing to 0.1% or more of the total relative dose were selected for inclusion and evaluation in all episodic releases. The screening calculation also yielded several short-lived radionuclides (without existing screening factors) with the potential to be important contributors to dose to exposed individuals based on the initial evaluation using the surrogate screening factors. To further examine the potential importance of these short-lived radionuclides without existing screening factors, we applied a less conservative and more realistic estimated screening factor. For this evaluation, we selected those radionuclides without existing screening factors and contributing to 1% or more of the total relative dose. For the shortest decay time of 20 minutes, 10 radionuclides met these criteria (^{89}Kr, ^{90}Rb, ^{90m}Rb, ^{93}Sr, ^{102}Mo, ^{102}Tc, ^{137}Xe, ^{139}Cs, ^{146}Pr and ^{146}Ce). No additional radionuclides were selected based on evaluation of the other decay times as the quantities of short-lived radionuclides of potential importance diminish rapidly with increasing decay times.

For these 10 radionuclides, we made several assumptions to estimate more realistic screening factors for use in place of the surrogate screening factors that were initially used. The estimated screening factors for the radionuclides without existing NCRP screening factors were based on radionuclides with existing screening factors, using half-life and beta energy as a guide for comparison to other radionuclides of the same isotope. This approach was taken because the inhalation and immersion doses (i.e., those most important for evaluation of these short-lived radionuclides) are likely primarily dependent on the absorption of beta particle energy. While not a perfect assumption, it is a reasonable one that allows for screening factor estimates to be relatively easily computed. The existing screening factor was then scaled according to the relative maximum beta particle energies for the two radionuclides. For example, the estimated screening factor for ^{102}Mo (which does not have an existing screening factor) was calculated using the screening factor for ^{101}Mo, scaled by the ratio of maximum beta energies (1.2 MeV for ^{102}Mo and 2.2 MeV for ^{101}Mo. This screening factor estimating process was used for 9 of the 10 radionuclides selected for this evaluation. The estimated screening factors for ^{90}Rb were used for the remaining radionuclide, ^{90m}Rb. Table 13 shows half-lives and maximum beta particle energies for the 10 radionuclides selected for this evaluation.

The relative doses for the 233 radionuclides were again calculated, using the estimated screening factors for the 10 radionuclides. With the exception of ^{93}Sr and ^{146}Ce, the estimated screening factors were lower than the initially assumed surrogate screening factors. Table 14 lists estimated screening factors for the 10 radionuclides without existing screening factors that were selected for inclusion and evaluation in all episodic releases.

Table 13. Half-Life Values and Maximum Beta Particle Energies for Selected Radionuclides

Radionuclide without existing screening factor	Half-life	Maximum beta energy (MeV)	Radionuclide with existing screening factor	Half-life	Maximum beta energy (MeV)
^{89}Kr	3.2 min	4.0	^{87}Kr	76 min	3.8
^{90}Rb	2.9 min	6.6	^{89}Rb	15 min	3.9
^{90m}Rb	4.3 min	6.6	^{90}Rb[a]	2.9 min	6.6
^{93}Sr	8.3 min	2.9	^{92}Sr	2.7 h	0.6
^{102}Mo	11.5 min	1.2	^{101}Mo	14.6 min	2.2
^{102}Tc	5.3 sec	4.4	^{104}Tc	18.2 min	3.0
^{137}Xe	3.8 min	4.1	^{89}Kr[b]	3.2 min	4.0
^{139}Cs	9.5 min	4.0	^{138}Cs	32.2 min	3.4
^{146}Pr	24.2 min	3.7	^{147}Pr	12 min	2.1
^{146}Ce	14 min	0.7	^{143}Ce	33 h	1.3

[a] The surrogate screening factors we calculated for ^{90}Rb were used to estimate screening factors for ^{90m}Rb
[b] The surrogate screening factors we calculated for ^{89}Kr were used to estimate screening factors for ^{137}Xe

Table 14. Estimated Screening Factors for Short-Lived Radionuclides with no Existing Atmospheric Screening Factor in NCRP (1996)

Radionuclide without existing screening factor	Radionuclide with existing screening factor	Scaled[a] Inhalation Screening Factor	Scaled Immersion Screening Factor	Scaled total screening factor for radionuclides in column 1
^{89}Kr	^{87}Kr	0	1.1E-6	1.1E-6
^{90}Rb	^{89}Rb	1.2E-7	3.2E-6	4.6E-6
^{90m}Rb	b	1.2E-7	3.2E-6	4.6E-6
^{93}Sr	^{92}Sr	7.3E-6	7.7E-6	3.8E-5
^{102}Mo	^{101}Mo	3.7E-8	7.1E-7	1.0E-6
^{102}Tc	^{104}Tc	2.2E-7	2.9E-6	4.4E-6
^{137}Xe	c	0	1.1E-6	1.1E-6
^{139}Cs	^{138}Cs	2.5E-7	3.1E-6	5.3E-6
^{146}Pr	^{147}Pr	7.9E-8	1.3E-6	1.9E-6
^{146}Ce	^{143}Ce	4.4E-6	1.8E-7	3.5E-5

[a] Scaled by the relative maximum beta energies shown in Table 13
[b] The estimated screening factor for ^{90}Rb was used
[c] The estimated screening factor for ^{89}Kr was used

Sustained Operation

For the sustained operation, we calculated a fission product inventory using the previously described reactor operating parameters (Table 12). We again selected release fractions of 0.1 for solids, 0.5 for halogens, and 1.0 for noble gases and deleted all radionuclides with zero inventory at a decay time of 40 days. Using the NCRP screening factors, we calculated relative screening

values for each radionuclide and sorted the list from highest to lowest, based on each radionuclide's percentage of the total cumulative relative dose. Beginning with the radionuclide with the highest percentage, we selected those radionuclides accounting for a cumulative 99.99% of the total relative dose for inclusion and evaluation in all episodic releases.

We developed our final list of radionuclides by combining all radionuclides selected after completing the above-described procedures. We also included any radionuclides selected by DOE (1991a) for evaluating the episodic releases that were not selected as part of our selection process. This resulted in the inclusion of ^{96}Nb and ^{129m}Xe, which likely were not selected by our process because of the different methodologies used for selection (i.e., DOE performed more detailed atmospheric modeling and calculated actual doses, whereas we relied on the use of simple screening factors). This process resulted in a list of 98 radionuclides (94 fission products; the activation product, ^{41}Ar; and the fuel element constituents, ^{234}U, ^{235}U, and ^{238}U) whose importance in terms of potential dose was assessed for all episodic release evaluations.

Episodic Release Evaluation Methodology

Several episodic events resulted in the release of known quantities of specific radionuclides to the atmosphere, including the Controlled Environmental Release Test (CERT), the Experimental Cloud Exposure Study (EXCES), and the Relative Diffusion Test (RDT) releases. Such episodic releases are relatively simple to screen or evaluate in terms of potential dose to exposed individuals. We can apply screening factors (NCRP 1996) directly to these known quantities to evaluate their relative importance with respect to other episodic releases.

Other episodic events resulted in the release of unknown quantities of many different radionuclides, including the Fuel Element Burn Test (FEBT), Fission Product Field Release Test (FPFRT), Initial Engine Test (IET), and Idaho Chemical Processing Plant (ICPP) criticality releases. This type of episodic release is considerably more difficult to screen because the precise quantity and composition of the release are not known. Because screening factors are only useful if they can be applied to known quantities of specific radionuclides, it is necessary to reconstruct the episodic event and estimate the radionuclides that were likely present during the release. This process involves estimating both the composition of radionuclides that may have been present during the test or accident and what fraction of each radionuclide may have actually been released to the environment. Once this has been completed, the NCRP (1996) screening factors can be used to evaluate the releases in terms of their relative importance with respect to other episodic releases.

For events that involved the release of short-lived radionuclides, we provide a list of ranked radionuclides and estimate the relative importance at different decay times. Considering decay time is essential because the importance of released radionuclides to potential dose changes as the short-lived radionuclides decay, and it allows us to rank these types of releases for different exposure scenarios. We ranked the releases at different decay times, depending on the location(s) of probable exposure based on the existing meteorological conditions. We also evaluated each release according to the transit time to the site boundary calculated by DOE (1991a). It was not clear whether the transit times reported by DOE (1991a) represent transit to the site boundary itself or to the perimeter location with the highest calculated dispersion factor; for the purposes of our evaluation, we assumed the latter. Events that involved the release of longer-lived radionuclides from aged fuel elements, such as the FEBT and FPFRT experiments, are not

sensitive to the relatively short decay time resulting from transit to the Site boundary so this was not an important parameter to consider.

We also must consider downwind dispersion to evaluate the overall importance of each given release at each location of probable exposure. We can use the basic Gaussian plume model to estimate downwind air concentrations for each radionuclide (Equation 1).

$$X(x,y,z) = \frac{Q}{2\pi\sigma_y\sigma_z u}\exp\left[-\frac{1}{2}\left(\frac{y}{\sigma_y}\right)^2\right]\left\{\exp\left[-\frac{1}{2}\left(\frac{z-H}{\sigma_z}\right)^2\right]+\exp\left[-\frac{1}{2}\left(\frac{z+H}{\sigma_z}\right)^2\right]\right\} \quad (4)$$

where

$X(x,y,z)$	= air concentration at coordinates x, y, and z from the source (Ci m^3)
x	= distance downwind from the source (m)
y	= perpendicular distance from the plume centerline or x-axis (m)
z	= distance or height above the ground surface, or xy plane (m)
Q	= source term (Ci s^{-1})
u	= mean wind speed at height H (m s^{-1})
δ_y	= horizontal dispersion coefficient at coordinate × (m)
δ_z	= vertical dispersion coefficient at coordinate × (m)
H	= effective plume height (m); source height plus plume rise

This equation assumes that the plume is reflected from the ground surface, i.e., there is no deposition at the surface. This is a conservative assumption because there will be some deposition of the radionuclides in the plume as they contact the ground surface.

We can make some additional assumptions to simplify Equation (4). We are concerned with ground-level concentrations (i.e., z = 0), resulting in Equation (5).

$$X(x,y,0) = \frac{Q}{\pi\sigma_y\sigma_z u}\exp\left[-\frac{1}{2}\left(\frac{y}{\sigma_y}\right)^2\right]\exp\left[-\frac{1}{2}\left(\frac{H}{\sigma_z}\right)^2\right] \quad (5)$$

For a ground level air concentration along the plume centerline (i.e., y and z = 0), Equation (5) simplifies further to Equation (6).

$$X(x,0,0) = \frac{Q}{\pi\sigma_y\sigma_z u}\exp\left[-\frac{1}{2}\left(\frac{H}{\sigma_z}\right)^2\right] \quad (6)$$

And finally, if we assumed no effective plume rise (i.e., H = 0), Equation (3) reduces to Equation (7).

$$X(x,0,0;H=0) = \frac{Q}{\pi\sigma_y\sigma_z u} \quad (7)$$

Equation (7) is used to address the downwind dispersion of the contaminant plume that occurs as it travels in the downwind direction away from the source. This equation is applicable to effluent releases on the order of a few minutes to a few hours, also frequently referred to as continuous point sources, and yields a calculated downwind concentration (i.e., curies per cubic meter [Ci m^{-3}]) if Q is expressed in curies per second. The equation also can be applied to instantaneous puff sources and yields the downwind total integrated concentration (Ci-s m^{-3}) when Q is expressed in curies. To maintain a consistent methodology for comparing the different episodic releases, and because the duration of each release event was not precisely known and varies considerably among the different releases, we have elected to evaluate the downwind dispersion using the total integrated concentration..

Using the total integrated concnetration presents a technical problem because several episodic events have resulted in releases occurring over periods longer than 1 day. Longer release times are typical for many of the ANP releases related to the IET runs, as well as ^{131}I releases resulting from the SL-1 accident. We have elected to evaluate releases for each episodic event assuming that the entire release from a given event occurs essentially at one time, even though many events involved releases occurring over a period of several days and even months. This approach is taken primarily to reduce the complexity involved with evaluating some of the release events. The implications of this methodology are addressed further in the final section on episodic releases that discusses results and provides recommendations.

The assumed wind speed in the equation is an important parameter because it influences both the quantities of radionuclides present (by impacting the time for decay) and the amount of atmospheric dispersion or dilution en route to the location of exposure. A higher wind speed increases the amount or quantity of some shorter-lived radionuclides by decreasing the time for decay, but it also generally decreases the downwind air concentrations by increasing atmospheric dispersion. Wind speed, time of day, and time of year also play important roles in determining the appropriate atmospheric stability class for each release.

We based our evaluations on existing meteorological conditions wherever possible; however, this information was sometimes not readily available. If it was known, the ground-level wind speed was used. Otherwise, the wind speed measured at the level nearest the ground was used. If existing wind speeds were not readily available for a given release, they were based on transit times calculated and reported by DOE (1991a). For releases not evaluated by DOE (1991a), the wind speed was assumed to be 2.0 m s^{-1}, a conservative value (NCRP 1996).

If existing wind speeds were readily available for a given release, stability classes were selected based on information provided by Clawson et al. (1989, Table VI-1). Unless the amount of cloud cover was specifically known, we assumed the most conservative stability class (i.e., the most stable class resulting in the least amount of atmospheric dispersion) for a given date, time, and wind speed. For releases not evaluated by DOE (1991a) that were known to occur on a specific day, we conservatively assume stable atmospheric stability (stability class F) for the purpose of selecting horizontal and vertical dispersion coefficients. Similarly, we have assumed neutral atmospheric stability for releases occurring over a period of more than a few days to reflect the fact that stable conditions were unlikely to remain constant for long periods of time.

For stability class D, Equations (8) and (9) were used to estimate the horizontal and vertical dispersion coefficients, respectively (NCRP 1996).

$$\sigma_y = \frac{0.08x}{\sqrt{1+0.0001x}} \quad (8)$$

$$\sigma_z = \frac{0.06x}{\sqrt{1+0.0015x}} \quad (9)$$

For stability classes other than D, horizontal and vertical dispersion coefficients were based on information provided by Clawson et al. (1989, Figures VI-7 and VI-8)

There are a number of reasons that using Equation (7) resulted in a conservative estimate of downwind dispersion. As previously mentioned, it assumed no deposition of contaminants on the ground surface. It also assumed that the contaminant plume traveled directly in a straight line at a constant speed to the nearest downwind person or point of potential exposure. In other words, these equations were short-term and were valid only for a period of time during which meteorological conditions remained reasonably constant. This is often not the case, as changes in wind speed and direction can result in additional downwind dispersion not accounted for by Equation (6). However, for evaluating the potential importance of short-term or episodic releases, these equations were appropriate and were simple to use in that they did not require detailed wind rose data. And finally, Equation (7) assumed no effective plume rise. This assumption tended to over estimate the downwind concentration at distances near the source because it did not account for the time and distance required for the contaminant plume to reach the ground surface if it has been released some distance above the ground surface. In fact, most episodic releases at the INEEL originated either from a stack or involved elevated temperatures causing some amount of plume rise.

To evaluate the releases, a relative screening value was calculated for each radionuclide using the same methodology described previously for selecting the radionuclides that are likely the most important contributors to dose. The primary difference was that we evaluated estimated quantities of radionuclides specific to each release, and we accounted for downwind dispersion. The relative screening value for each radionuclide was calculated using Equation (10).

$$RV = X \cdot SF \quad (10)$$

where
RV = relative screening value (unitless)
X = estimated downwind total integrated concentration (assumed unitless)
SF = NCRP atmospheric screening factor (assumed unitless)

A cumulative relative screening value was then calculated for each episodic release by summing the relative screening values for all radionuclides present for that release event. The episodic releases were finally prioritized according to potential dose to an exposed individual based on the cumulative relative screening value for each release event. Exposure at onsite locations was evaluated using inhalation and plume immersion screening factors, while exposure at offsite locations was evaluated using total screening factors. As mentioned previously for the radionuclide selection process, we used the screening factors for the parent radionuclide only, excluding the contribution from daughter products given by NCRP (1996).

It is important to understand that these screening values do not result in an estimated dose to an exposed individual, but they can be used to determine which episodic releases are likely to be most important in terms of potential dose to an exposed individual. This enables time and resources to be focused on these events. Those release events determined to be the most important may require additional detailed calculations of actual dose to determine their overall importance with regard to annual releases at the INEEL, such as those that have occurred routinely at ICPP or TRA.

Episodic Release Evaluations

The following episodic release events at the INEEL were evaluated based on the methodology described above. Each evaluation considered the list of 98 radionuclides determined to be the most important contributors to potential dose. Cumulative screening values for each release are provided in individual Excel spreadsheets, which can be accessed using the hyperlink toward the end of each release evaluation. For most releases, screening values are provided for both offsite exposures (based on total screening factors) and onsite exposures (based on inhalation and plume immersion screening factors only), where appropriate. Figure 1 shows an INEEL site map and includes the locations of the various site facilities and projects.

Meteorological conditions existing at the time of each release as well as the offsite locations of highest exposure and transit times determined by DOE (1991a) were used to estimate decay times and locations of probable exposure. The nearest offsite locations used for this analysis are shown in Figure 21 and are the same 16 locations assumed by DOE (1991a). Most of these locations likely represent occupied residences during the times of most of the releases. Three of the locations, however, probably do not represent occupied residences. These include Cerro Grande, an abandoned railroad stop; Frenchman's Cabin, an abandoned cabin near Big Southern Butte; and Cellar, a potato cellar southeast of the INEEL Site.

BORAX-I Excursion (July 22, 1954)

The BORAX-I excursion involved a water-cooled, water-moderated reactor that was operated during nondestructive experiments in the latter part of 1953 and the early summer of 1954. The experiments were carried out at the ANL-W facility, and a final destructive excursion was planned following the completion of the nondestructive tests. This final excursion was expected to result in the melting of some portion of the fuel elements, and fallout plates and film were positioned to evaluate the environmental impacts of the released fission products.

After a day of waiting for favorable wind conditions, the destructive test was conducted on the morning of July 22, 1954. Shortly before 8:00 a.m., the U.S. Weather Bureau notified ANL officials that conditions were favorable for beginning the experiment. At 8:20 a.m., the central rod was ejected from its fully inserted position, and shortly after, a column of dark gray smoke was ejected from the reactor to a height of approximately 80 feet. Monitoring teams dispatched shortly after the excursion determined the trajectory of the cloud to have been roughly in a southwesterly direction, passing over Adams Boulevard approximately 1.5 miles from the Van Buren-Adams junction and continuing over the Union Pacific Railway in the same direction. Construction personnel working in the vicinity of the ZPPR, which is approximately 0.6 miles from the BORAX reactor, were immediately evacuated, and traffic control was established on

U.S. Highway 20 and on Van Buren Avenue leading to EBR (Brodsky and Beard 1960). Dietrich (1954) reported that the air-dispersed material was blown in a direction about 35 degrees west of south. The wind was reportedly blowing from the northeast with a speed of 8 mph at ground level and 20 mph at the 250-foot level.

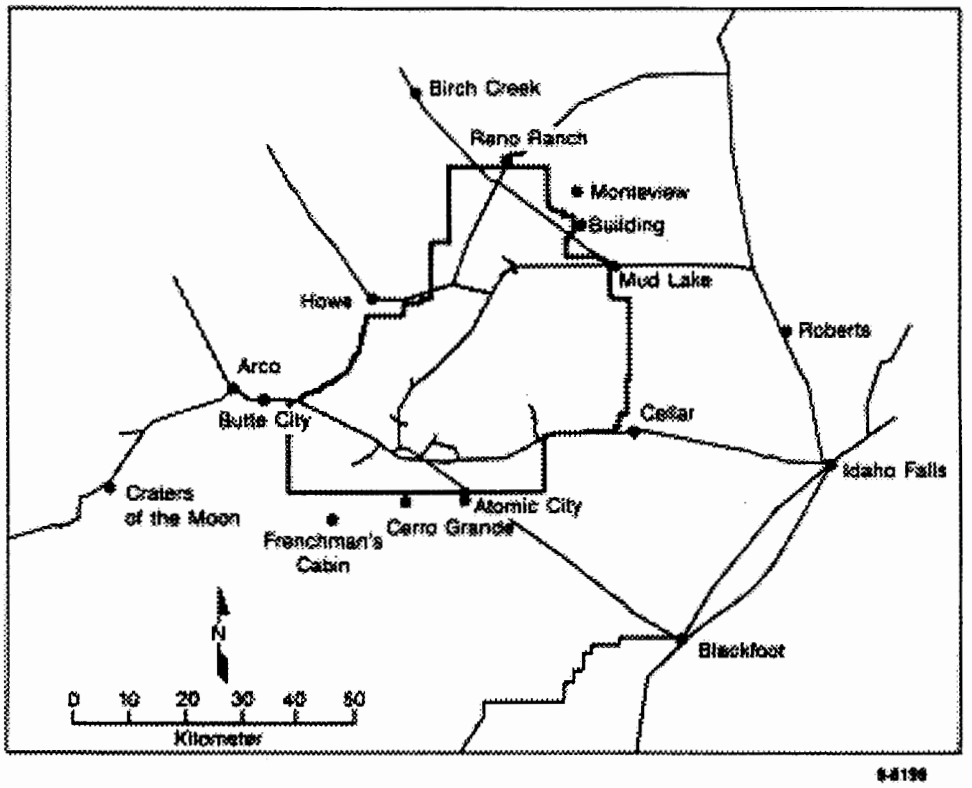

Figure 21. Map showing 16 offsite locations at which potential exposure was assessed (DOE 1991a).

Based on the reactor operating parameters provided by Dietrich (1954) and DOE (1991a), we used the RSAC-5 computer code to calculate the expected fission product inventory for the BORAX reactor core. The reactor was operated a number of times during 1953 and 1954 before the destructive test of July 22, 1954. To estimate the fission product inventory in the reactor at the time of the test, three simplified operation periods were assumed: 550 MW for 1 second on December 30, 1953; 25,000 MW for 1 second on December 31, 1953; and 280 MW for 1 second on June 30, 1954. The destructive test was assumed to produce 52,000 MW of energy during an operating period of 0.0026 second, resulting in a total energy release of 135 MW-s.

Assumed release fractions for the BORAX-I destructive test were based on the release fractions determined for the SPERT-I, Test No. 1. Both of these tests employed plate type elements and resulted in substantial destruction of the core. The energy release for the BORAX-I test was approximately a factor of 4 greater than that for the SPERT-I, Test No. 1. Using the same

assumptions made by DOE (1991a), we increased the assumed SPERT-I noble gas release fraction by a factor of 4, and increased the halogen release fraction by a factor of 4 as well as an additional factor of 10 to ensure a conservative estimate (i.e., a total increase of 40 times for halogens). Finally, we selected a release fraction for fission product solids and uranium that was one-fourth the fraction selected for the halogens. Based on this information, we assumed release fractions of 0.4 for noble gases, 0.004 for halogens, and 0.001 for solids and uranium in the fuel. Dietrich (1954) reports that each fuel element contained 18 fuel plates with a combined ^{235}U content (90% enriched) of about 140 grams, so the 30 elements in the reactor at the time of the test contained approximately 4200 grams ^{235}U.

We assumed a wind speed of 8 mph (3.6 m s^{-1}), atmospheric stability class D, and a straight-line trajectory to the nearest offsite location in the general direction of the prevailing winds, considering the location with the highest dispersion factor reported by DOE (1991a). These assumptions resulted in the plume passing over Highway 20 (onsite, 10 km downwind) approximately 46 minutes following the release and arriving at Atomic City (offsite, 20 km downwind) after approximately 93 minutes. DOE (1991a) reported a transit time to the INEEL site boundary of 0.9 hours (54 minutes), which corresponded to a wind speed of 6.2 m s^{-1}. Therefore, we also evaluated exposure to the plume in Atomic City 54 minutes following the release, assuming a wind speed of 6.2 m s^{-1}. Based on these decay times and travel distances, the relative screening value for the BORAX-I release for the onsite exposure was 2.7×10^{-9} and for the offsite exposure was 2.2×10^{-9} for the 93 minute transit time and 1.6×10^{-9} for the 54 minute transit time.

NRF S1W Engineering Test

Engineering experiments were conducted on the first S1W prototype reactor core at the NRF from June 18 through July 1, 1955. The test was designed to examine the limits of fuel element performance beyond operating limits and was important in the development of subsequent naval reactor designs. During testing on June 18, a small portion of the assembly released fission products to the primary cooling system. The test was continued through June 30, 1955, after which the test assembly was removed from the reactor for examination (Bradley 1991).

The release was evaluated using effluent data provided by Bradley (1991). These data were also used by DOE (1991a) and subsequently decayed according to a 3.7-hour transit time before reaching the nearest offsite person. To conservatively evaluate this release, we elected to use the non-decayed data (i.e., quantities measured directly at the point of release) for our analysis.

We assumed a wind speed of 4.5 mph (2.0 m s^{-1}), atmospheric stability class D, and a straight-line trajectory to the nearest offsite location in the general direction of the prevailing winds, based on the transit time and location with the highest dispersion factor reported by DOE (1991a). These assumptions resulted in the plume passing over Highway 33 (onsite, 30 km downwind) and the Building location (offsite, 40 km downwind). DOE (1991a) reported a transit time to the INEEL site boundary of 3.7 hours (222 minutes), which corresponded to a wind speed of 3.0 m s^{-1}. Therefore, we also evaluated exposure to the plume in Atomic City, assuming a wind speed of 3.0 m s^{-1}. Based on these wind speeds and travel distances, the relative screening value for the NRF S1W Engineering Test release for the onsite exposure was 1.1×10^{-9} and for the offsite exposure was 5.1×10^{-8} for the 2.0 m s^{-1} wind speed and 3.4×10^{-8} for the 3.0 m s^{-1} wind speed.

Initial Engine Tests of the ANP Program (1955-1961)

The ANP Program was designed to investigate the feasibility of developing a nuclear propulsion system for aircraft of unlimited range for military use. The test series were conducted at the NRTS (now the INEEL) by the Aircraft Nuclear Propulsion Department of the General Electric Company under contracts with the U.S. Air Force and AEC (Thornton et al. 1962, 51389). There were a total of 26 Initial Engine Test (IET) runs involving three separate reactor assemblies, designated Heat Transfer Reactor Experiment (HTRE) No. 1, 2, and 3. The program continued through March 28, 1961, when it was canceled.

The tests were conducted at the CTF, located at the TAN area. The HTRE reactor assemblies were mounted on a four-track railroad dolly, allowing for operation of only one HTRE assembly at a time. The CTF provided the shielded control room, the support utilities required for testing, and the instrumented reactor exhaust system.

Thornton et al. (1962) provides a general description of the HTRE-1 reactor assembly. Three reactor power operations were conducted using the HTRE-1 assembly, including IETs #3, #4, and #6. The reactor operated at power levels up to 20.2 MW and generated about 5500 MW-h of nuclear energy. Initial Engine Tests #1, #2, and #5 did not involve power operations of the reactor, and consequently did not involve atmospheric releases.

Flagella (1962) provides a general description and summary of the HTRE-2 reactor assembly. Evans (1957) provides additional information related to the HTRE-2 reactor assembly and associated engine tests. Miller et al. (1960) and GE (1959) provide details related to the HTRE-3 reactor assembly and associated engine tests.

The various test series involved a number of power reactor operations that resulted in the release of radionuclides to the environment. The dates and times of reactor operations, and consequently atmospheric releases, varied throughout the duration of the project. Therefore, the meteorological conditions that existed during the tests depended on the time of the tests. All operations were under strict meteorological control, and permissible wind directions seriously limited operations. On many days, it was impossible to operate at all, and most of the time operation was possible only a few hours each day (Thornton et al. 1962).

To focus our efforts on the potentially most important test series, we evaluate the 26 IET runs in the order of the numbered tests, with greatest emphasis on the tests that involved the largest offsite dose calculated by DOE (1991a). Because of the complexity related to the reconstruction of releases from these tests and the often discontinuous reactor operations, we made a number of simplifying assumptions, discussed specifically for each test. In addition, DOE (1991a) expended a substantial amount of effort in reconstructing these releases. Again, because of the complicated nature of reconstructing releases from these tests, we have relied on some of the work previously completed and reported by DOE (1991a). However, in all cases, we carefully evaluated the procedures used and assumptions made to ensure an appropriate and conservative evaluation. Additionally, in some cases, we based our estimates of releases on a more simplified treatment of events than that reported by DOE (1991a).

IET-1

Thornton et al. (1962) reports that IET-1 was not a power operation. As such, it did not involve any radioactive releases.

IET-2

Thornton et al. (1962) reports that IET-2 was not a power operation. As such, it did not involve any radioactive releases.

IET-3

The first test series involving power operations covered the period from December 27, 1955, to February 25, 1956, and was designated IET No. 3 (Thornton et al. 1962). The core used in this test series was called the A2 core and was part of the first test assembly, the D101A2. Studies during IET-3 were designed to observe reactor and engine behavior during both chemically assisted and full nuclear operation.

Release of radioactive material as a burst of stack activity was first detected on February 11, 1956, during an attempted transfer to full nuclear power. The presence of fission fragments was established during subsequent operation by the detection of ^{131}I in the stack gas. Fuel cartridge damage was suspected and later verified, during disassembly of the A2 core, as the cause and ranged from ring buckling to burning and melting. Two cartridges were severely damaged, while only one other showed any melting or burning.

We calculated an estimated fission product inventory, based on the operational history provided by Thornton et al. (1962), using the RSAC-5 computer code. We assumed release fractions of 0.1, 0.5, and 1.0 for solids, halogens, and noble gases, respectively, in the two severely damaged cartridges. In terms of the total core inventory, which consisted of 37 fuel cartridges, this equates to overall release fractions of 0.0054, 0.027, and 0.054 for solids, halogens, and noble gases, respectively. Thornton et al. (1962) reports that the total uranium in the core amounted to 90 pounds, or 40,800 grams. We assumed the uranium in the core was released in the same fraction as that used for the solids. We assumed production and consequent release of 2250 Ci ^{41}Ar, as reported by DOE (1991a).

Thornton et al. (1962) report activity releases of 2000 Ci during a 4-hour period and 1000 Ci during a 2-hour period during 100% nuclear operation, as well as 100 Ci during a series of tests on the last day of operation. These data suggest a 500 curies per hour (Ci hr^{-1}) release rate during 100% nuclear operation. Assuming this rate of release during the entire 40 hours of operation above 200 kW, a total release of 20,000 Ci was obtained. Additionally, a peak release rate of approximately 30 Ci hr^{-1} particulate activity is reported for IET-3. Assuming this rate of release during the entire 40 hours of operation, a total release of 1200 Ci was obtained. Our reconstructed operation resulted in a total release of 39,300 Ci of 10-minute decayed particulate activity[10], or a

[10] This release amount accounts only for the 94 selected fission products evaluated for each release; the actual total release amount for all produced fission products would be greater. Throughout the IET analyses (with the exception of IET-10), we conservatively make comparisons to reported releases in this manner.

release rate of nearly 1000 Ci hr^{-1} during the 40 hours of operation, which is conservative by comparison to either reported release rate.

We assumed a wind speed of 10.9 mph (4.9 m s^{-1}), atmospheric stability class D, and a straight-line trajectory to the nearest offsite location in the general direction of the prevailing winds, based on the transit time and location with the highest dispersion factor reported by DOE (1991a). These assumptions resulted in the plume passing over Highway 28 (onsite, 11 km downwind) approximately 38 minutes following the release and arriving at the Building (offsite, 14 km downwind) after approximately 48 minutes. Based on these decay times and travel distances, the relative screening value for the IET-3 release for the onsite exposure was 1.5×10^{-7} and for the offsite exposure was 5.6×10^{-6}.

IET-4

The IET-4 test series was conducted between April 17 and June 29, 1956 and employed an A2 core to which several significant repairs and modifications were made as a result of IET-3 operations. Thirteen new fuel cartridges with extra rails were installed, and fifteen control rods were replaced. The primary purpose of this test series was to determine whether modifications based on the results of the first test series (IET-3) had significantly improved the capabilities of the reactor (Thornton et al. 1962).

Data regarding releases during this test series are somewhat limited, but a number of tests were performed in an attempt to correlate exhaust-gas activity to power level, fuel flow, and plate temperature. It was concluded that the plate temperature level was by far the most critical parameter influencing the release of particulate activity, which showed a sharp increase at the highest tested plate temperatures. Post operation evaluation of the fuel cartridges revealed three severely damaged cartridges.

We calculated an estimated fission product inventory, based on the operational history provided by Thornton et al. (1962), using the RSAC-5 computer code. We assumed release fractions of 0.1, 0.5, and 1.0 for solids, halogens, and noble gases, respectively, in the three severely damaged cartridges. In terms of the total core inventory, which consisted of 37 fuel cartridges, this equated to overall release fractions of 0.0081, 0.041, and 0.081 for solids, halogens, and noble gases, respectively. Thornton et al. (1962) reports that the total uranium in the core amounted to 90 pounds, or 40,800 grams. We assumed the uranium in the core was released in the same fraction as that used for the solids. We conservatively assumed ^{41}Ar to be released at a rate of 56 Ci hr^{-1} of operation at a power level above 200 kW, as reported by DOE (1991a).

Particulate release rates ranging from 16 to 186 Ci hr^{-1} were reported for the tests conducted to examine particulate activity as a function of plate temperature. Conservatively assuming a 186 Ci hr^{-1} release rate during the entire 194 hours of operation above 200 kW resulted in a total particulate release of 36,084 Ci. Additionally, Thornton et al. (1962) reports a peak release rate of approximately 2 Ci hr^{-1} particulate activity for IET-4. Assuming this rate of release during the entire 194 hours of operation, a total release of 388 Ci was obtained. Our reconstructed operation resulted in a total release of 153,000 Ci of 10-minute decayed particulate activity, or a release rate of nearly 700 Ci hr^{-1} during the 194 hours of operation, which is conservative by comparison to either reported release rate.

DOE (1991a) evaluated IET-4 releases during three separate periods; as such, three different transit times were reported. Because the calculated relative screening value generally increases with increasing transit time, we elected to evaluate the IET-4 releases using the longest reported transit time. Consequently, we assumed a wind speed of 7.9 mph (3.5 m s^{-1}) and a straight-line trajectory to the nearest offsite location in the general direction of the prevailing winds, based on the transit time and location with the highest dispersion factor reported by DOE (1991a). These assumptions resulted in the plume passing over Highway 28 (onsite, 11 km downwind) approximately 52 minutes following the release and arriving at the Building (offsite, 14 km downwind) after approximately 66 minutes. Based on these decay times and travel distances, the relative screening value for the IET-4 release for the onsite exposure was 9.6×10^{-7} and for the offsite exposure was 3.8×10^{-5}.

IET-5

Thornton et al. (1962) reports that IET-5 was a series of shielding tests, which apparently did not involve power operation of the reactor. As such, it did not involve radioactive releases.

IET-6

The IET-6 test series was performed from September 24, 1956 through January 3, 1957 and employed a completely new reactor test assembly called the D101A3 (Thornton et al. 1962). This was the last test series that used the HTRE-1 assembly. The new A-3 core assembly differed from the A-2 core used during IET-3 and IET-4 in that the new insulation sleeve was designed to enhance the structural integrity of the sleeve against pressure collapse. This design resulted from an intensive development effort performed to determine the cause of fuel cartridge damage and to prepare for operation of the A3 reactor (Thornton et al. 1962).

We calculated an estimated fission product inventory, based on the operational history provided by Thornton et al. (1962), using the RSAC-5 computer code. The release rate during IET-6 was considerably less than that during IET-3, and less extensive fuel cartridge damage occurred. Therefore, we assumed release fractions of 0.1, 0.5, and 1.0 for solids, halogens, and noble gases, respectively, in a single fuel cartridge. In terms of the total core inventory, which consisted of 37 fuel cartridges, this equated to overall release fractions of 0.0027, 0.014, and 0.027 for solids, halogens, and noble gases, respectively. Thornton et al. (1962) reports that the total uranium in the core amounted to 90 pounds, or 40,800 grams. We assumed the uranium in the core was released in the same fraction as that used for the solids. We conservatively assumed ^{41}Ar to be released at a rate of 56 Ci hr^{-1} of operation at a power level above 200 kW, as reported by DOE (1991a).

The first indication that some amount of damage had occurred within the A-3 core was detected on the night of December 18, 1956. A number of measurements were made to assess the activity release rate as a function of reactor power level, and the data consistently suggest higher release rates at increased power levels, with a maximum reported release rate of 25 Ci hr^{-1}. Conservatively assuming a 25 Ci hr^{-1} release rate during the entire 254 hours of operation above 200 kW resulted in a total particulate release of 6,350 Ci. Additionally, Thornton et al. (1962) reports a peak release rate of less than 1 Ci hr^{-1} particulate activity for IET-6. Assuming this rate of release during the entire 254 hours of operation, a total release of less than 254 Ci was

obtained. Our reconstructed operation resulted in a total release of 13,600 Ci of 10-minute decayed particulate activity, or a release rate of about 54 Ci hr^{-1} during the 254 hours of operation, which is conservative by comparison to either reported release rate.

We assumed a wind speed of 4.5 mph (2.0 m s^{-1}) and a straight-line trajectory to the nearest offsite location in the general direction of the prevailing winds, based on the transit time and location with the highest dispersion factor reported by DOE (1991a). These assumptions resulted in the plume passing over Highway 20 (onsite, 35 km downwind) approximately 285 minutes following the release and arriving at Atomic City (offsite, 42 km downwind) after approximately 342 minutes. Based on these decay times and travel distances, the relative screening value for the IET-6 release for the onsite exposure was 8.2×10^{-8} and for the offsite exposure was 6.7×10^{-6}.

IET-7

The IET-7 tests consisted of a series of critical experiments (i.e., the reactor was made critical, but at low power) to determine the nuclear characteristics of the HTRE-2 reactor assembly and at least two insert test mockups (DOE 1991a). The reactor operating times were short (20-minute intervals) and the power levels were very low (~6 W). Consequently, it is assumed that radionuclide releases to the environment were negligible relative to releases associated with other reactor power operation tests.

IET-8

Evans (1957) reports that the IET-8 test series was performed between July 18 and August 28, 1957. It was the first power operation of the HTRE-2 reactor assembly and involved evaluation of the insert 1-B. A detailed operational history was not provided, but it is indicated that a total of 33.97 hours of operation at maximum power had been accumulated during the test series. Additionally, it is noted that no fission fragment evolution was measured, but the presence of molybdenum, manganese, and other radioactive particles was indicated by detection on filter papers.

We calculated an estimated fission product inventory, based on the operational history provided by Evans (1957) and further refined by DOE (1991a), using the RSAC-5 computer code. We made additional conservative assumptions and set the average power during reactor operation to the maximum reported power level (11.8 MW). Although no fission product evolution was measured, we elected to conservatively assume that noble gases were released. Consequently, we assumed release fractions of 0, 0, and 1.0 for solids, halogens, and noble gases, respectively, in a single fuel cartridge. In terms of the total core inventory, which consisted of 30 fuel cartridges, this equated to overall release fractions of 0, 0, and 0.033 for solids, halogens, and noble gases, respectively. We assumed no release of uranium fuel constituents, which was consistent with the release of no particulate fission products.

We assumed the particulate release of 153 Ci between August 16 and 28 reported by Evans (1957) to be composed entirely of molybdenum and manganese activation products. We made this assumption because it was reported that no fission fragment evolution was measured, molybdenum and manganese particles were detected, and molybdenum and manganese deterioration of the outer cladding had occurred. Radionuclides that would be expected to result from the activation of molybdenum and manganese include ^{56}Mn, ^{93m}Mo, ^{93}Mo, ^{99}Mo, and ^{101}Mo.

We used the slightly higher release value of 243 Ci reported by DOE (1991a) and assumed the release consisted entirely of ^{93}Mo, the radionuclide with the highest total NCRP screening factor.

DOE (1991a) evaluated IET-8 releases during several separate periods; as such, variable transit times were reported. Because the calculated relative screening value generally increases with increasing transit time, we elected to evaluate the IET-8 releases assuming a relatively slow wind speed and a straight-line trajectory to the nearest offsite location. Consequently, we assumed a wind speed of 4.5 mph (2.0 m s^{-1}) and a straight-line trajectory to the Building location. These assumptions resulted in the plume passing over Highway 28 (onsite, 11 km downwind) approximately 92 minutes following the release and arriving at the Building (offsite, 14 km downwind) after approximately 117 minutes. Based on these decay times and travel distances, the relative screening value for the IET-8 release for the onsite exposure was 7.8×10^{-8} and for the offsite exposure was 6.1×10^{-7}.

IET-9

The IET-9 tests consisted of a series of critical experiments to determine the nuclear characteristics of the HTRE-2 reactor assembly, similar to those described for IET-7 (DOE 1991a). The critical experiments were performed with nuclear mockups of the test inserts, which were not designed for power operations, so the power levels during these tests were very low by comparison to other IET operations. Consequently, it is assumed that radionuclide releases to the environment were negligible relative to releases associated with other reactor power operation tests.

IET-10

Foster et al. (1958) reports that the IET-10 test series commenced on December 12, 1957 and was terminated on March 6, 1958. This test employed the HTRE-2 reactor assembly and utilized the insert 2B, the first in a series of ceramic insert tests. The program was divided into three phases (Phase I, II, and III), using the original, first modification (IET-14), and second modification (IET-18) of the orifice plate.

DOE (1991a) calculated estimated power levels for the various operating periods during IET-10, based on the insert fission rate reported in Table 1 of Foster et al. (1958). These power levels resulted in a total reactor power of approximately 140 MW-h. However, Foster et al. (1958) reports an accumulated power of approximately 50 MW-h during Phase I and approximately 550 MW-h during Phase II. The total power for Phase III was not reported by Foster et al. (1958), but DOE (1991a) reports a power level of 1897 MW-h for Phase III. Summing the accumulated power for each of the phases resulted in a total power of approximately 2500 MW-h for IET-10, which is significantly higher than the 140 MW-h resulting from the reactor operating history assumed by DOE (1991a). Therefore, we modified the power levels used by DOE (1991a) by a factor of 18, resulting in a total power of 2517 MW-h, and used these data to calculate a fission product inventory using RSAC-5. It appears likely that DOE (1991a) estimated the portion of total reactor power produced by the insert. To be conservative,

we have elected to assume that the total reactor power was produced by the test insert[11]. Additionally, assuming release of the entire fission product inventory resulting from a reactor power of 140 MW-h would not result in a total release that is conservative by comparison to the release rates discussed below.

Through an extensive amount of data analysis related to radioiodine release rates, DOE (1991a) estimated release fractions for each run completed as part of IET-10. We initially elected to use the highest release fractions (0.000653, 0.0043, and 0.653 for solids, halogens, and noble gases, respectively) estimated by DOE (1991a), but this resulted in a fission product release rate that was not conservative by comparison to release rates reported by Foster et al. (1958), discussed below. Therefore, we increased these release fractions in an iterative process to approximate a conservative estimate of total releases based on data reported by Foster et al. (1958), resulting in assumed release fractions of 0.075, 0.1, and 1.0 for solids, halogens, and noble gases, respectively. These release fractions are conservative by comparison to the calculated leakage rates reported by Foster et al. (1958), but we elected to base our release on the most conservative data available (i.e., the maximum reported release rates). We assumed the same uranium and ^{41}Ar release values estimated by DOE (1991a)[12].

There were three separate phases of testing during IET-10. Foster et al. (1958) reports a maximum fission product release rate of 2630 Ci hr^{-1} of 17-second decayed activity during Phase I. The highest release rate reported for Phase II is 1825 Ci hr^{-1}. Different release rates are reported for the various runs associated with the Phase III endurance testing, and we have applied the corresponding release rates for each run number. Where release rates for a particular run during Phase III are not reported, we have assumed the release rate reported for the previous run. Applying the reported rates of release during the 159 hours of assumed operation for the three phases of testing would result in a total release of over 6.7×10^6 Ci. Our reconstructed operation resulted in a total release of more than 6.9×10^6 Ci of 17-second decayed fission products, which is conservative by comparison to the most conservative reported release rate.

DOE (1991a) evaluated IET-10 releases during several separate periods; as such, variable transit times were reported. Because the calculated relative screening value generally increases with increasing transit time, we elected to evaluate the IET-10 releases assuming a relatively slow wind speed and a straight-line trajectory to the nearest offsite location. Consequently, we assumed a wind speed of 4.5 mph (2.0 m s^{-1}), atmospheric stability class D, and a straight-line trajectory to the Building location. These assumptions resulted in the plume passing over Highway 28 (onsite, 11 km downwind) approximately 92 minutes following the release and arriving at the Building (offsite, 14 km downwind) after approximately 117 minutes. Based on these decay times and travel distances, the relative screening value for the IET-10 release for the onsite exposure was 1.1×10^{-5} and for the offsite exposure was 4.0×10^{-4}.

[11] DOE (1991a) typically assumed fractional power production by the test inserts for the various IET tests. We have conservatively assumed that the total reported reactor power was produced by the test insert (except IET-12).

[12] After reviewing the methodology used by DOE (1991a) to estimate ^{41}Ar and uranium isotope releases for a number of tests and because these radionuclides are not important contributors to the relative ranking values, we concluded that the methodology is appropriate and have elected to assume the same values reported by DOE for this and several other releases.

IET-11

The IET-11 test series was conducted between March 12 and April 14, 1958 (Evans 1958). This test series employed the HTRE-2 reactor assembly and the insert 1-C (D101-C3). The insert 1-C was devised to evaluate the mechanical and materials characteristics of unclad, slotted, hydrided zirconium as a core neutron moderating material. Airborne radionuclide releases were documented to have occurred beginning on March 20, 1958, when the reactor was operated at power levels exceeding 120 kW.

We calculated an estimated fission product inventory, based on the operational history provided by Evans (1958), using the RSAC-5 computer code. Through an extensive amount of data analysis based on release rates reported by Baker et al. (1959), DOE (1991a) estimated release fractions for each run completed as part of IET-11. We initially selected the highest release fractions estimated by DOE (1991a), but this resulted in a fission product release rate that was not conservative by comparison to release rates reported by Flagella (1962). Therefore, we increased these release fractions by a factor of 20, resulting in assumed release fractions of 1.48×10^{-7}, 2.36×10^{-5}, 0.00114, and 0.00114 for solids, cesium isotopes, halogens, and noble gases, respectively. We assumed the same uranium and ^{41}Ar release values estimated by DOE (1991a).

Flagella (1962) reports that stack gas radiation monitoring equipment used during the test series indicated a fresh fission product (decayed 10 minutes) release rate of 14 Ci hr^{-1} when the reactor power was increased to produce a 700°F insert moderator temperature. Following this unexpected release, the core was returned to the hot shop for examination and cleaning. It was established that no fuel-cartridge rupture had occurred, but 8.4 grams of ^{235}U that was likely deposited during the insert 2B operation was flushed from the lower cocoon. When testing resumed, a release rate of 7.7 Ci hr^{-1} was measured. Conservatively assuming a 14 Ci hr^{-1} release rate during the entire 141 hours of operation above 120 kW resulted in a total particulate release of 1,974 Ci. Our reconstructed operation resulted in a total release of 3,540 Ci of 10-minute decayed activity, or a release rate of over 25 Ci hr^{-1} during the 141 hours of operation, which is conservative by comparison to either reported release rate.

DOE (1991a) evaluated IET-11 releases during several separate periods; as such, variable transit times were reported. Because the calculated relative screening value generally increases with increasing transit time, we elected to evaluate the IET-11 releases assuming a relatively slow wind speed and a straight-line trajectory to the nearest offsite location. Consequently, we assumed a wind speed of 4.5 mph (2.0 m s^{-1}), atmospheric stability class D, and a straight-line trajectory to the Building location. These assumptions resulted in the plume passing over Highway 28 (onsite, 11 km downwind) approximately 92 minutes following the release and arriving at the Building (offsite, 14 km downwind) after approximately 117 minutes. Based on these decay times and travel distances, the relative screening value for the IET-11 release for the onsite exposure was 3.6×10^{-8} and for the offsite exposure was 2.1×10^{-6}.

IET-12 (BOOT)

The IET-12 test series, also referred to as Operation Burnout One Tube (BOOT), was carried out between April 21 and May 7, 1958. This test series employed the HTRE-2 reactor assembly and the insert 1-D and was designed to ascertain the consequences of severely restricting the coolant airflow through an operating reactor. Releases of radioactive material occurred on May 2,

1958, when the airflow to one of six tubes in the insert 1-D (tube 6) was restricted during BOOT 1, melting a portion of the tube. A second attempt (BOOT 2) to burn out another tube was aborted on May 6, 1958, so the vast majority of the releases associated with this test series occurred on May 2.

We calculated an estimated fission product inventory, based on the operational history provided by Devens et al. (1958) and further refined by DOE (1991a), using the RSAC-5 computer code. Baker et al. (1959) reports that about 843 grams from a total of 2,000 grams (gross weight) in the damaged fuel cartridge (about 42%) could not be accounted for and was assumed to have passed along to or through the exhaust system. We conservatively assumed release fractions of 0.50 (50%) for solids, halogens, and noble gases in tube 6. The insert 1D generated about 11% of the total test power, so we conservatively assumed that each of the six tubes in the insert generated 5% of the total test power and modified the release fractions accordingly. In terms of the total reactor inventory, this resulted in an assumed release fraction of 0.025 for the entire fission product inventory. We assumed the same uranium and ^{41}Ar release values estimated by DOE (1991a).

Devens et al. (1958) reports a total fission product release of 21,000 Ci as indicated by the radiation monitor on the 76-inch effluent duct and a total release of 13,000 Ci as indicated by the stack monitor. Our reconstructed operation resulted in a total release of more than 43,400 Ci of 10-minute decayed fission products, which is conservative by comparison to either reported release.

DOE (1991a) evaluated IET-12 releases during several separate periods; as such, variable transit times were reported. Because the calculated relative screening value generally increases with increasing transit time, we elected to evaluate the IET-12 releases assuming a relatively slow wind speed and a straight-line trajectory to the nearest offsite location. Consequently, we assumed a wind speed of 4.5 mph (2.0 m s^{-1}), atmospheric stability class D, and a straight-line trajectory to the Building location. These assumptions resulted in the plume passing over Highway 28 (onsite, 11 km downwind) approximately 92 minutes following the release and arriving at the Building (offsite, 14 km downwind) after approximately 117 minutes. Based on these decay times and travel distances, the relative screening value for the IET-12 release for the onsite exposure was 6.6×10^{-8} and for the offsite exposure was 1.8×10^{-6}.

IET-13

The IET-13 test series consisted of the critical experiments and low power testing phase of the HTRE-3 reactor assembly. The testing was performed between September 8 and November 18, 1958. An unexpected nuclear excursion on November 18, 1958 resulted in the release of radioactive material to the environment. The critical experiments and low power testing did not produce significant airborne releases by comparison to the excursion.

We calculated an estimated fission product inventory, based on the operational history provided by Devens (no date), using the RSAC-5 computer code. DOE (1991a) estimated release fractions based on an extensive review of available data. We determined that these release fractions were appropriate and elected to use the same values, resulting in assumed release fractions of 0.000375, 0.0063, 0.0375, and 0.094 for solids, cesium isotopes, halogens, and noble gases, respectively. We assumed the same uranium and ^{41}Ar release values estimated by DOE (1991a).

Wilks (1959) reports a gross fission product release of 400 Ci. Our reconstructed operation resulted in a total release of more than 9,500 Ci of 10-minute decayed fission products, which is conservative by comparison to the reported release.

We assumed a wind speed of 11.1 mph (5.0 m s^{-1}), atmospheric stability class D, and a straight-line trajectory to the nearest offsite location in the general direction of the prevailing winds, based on the transit time and location with the highest dispersion factor reported by DOE (1991a). These assumptions resulted in the plume passing over Highway 22 (onsite, 10 km downwind) approximately 34 minutes following the release and arriving at Howe (offsite, 25 km downwind) after approximately 84 minutes. Based on these decay times and travel distances, the relative screening value for the IET-13 release for the onsite exposure was 5.5×10^{-9} and for the offsite exposure was 4.1×10^{-9}.

IET-14

The IET-14 test series was conducted between March 27 and May 20, 1959. Documented releases of activity began on April 17, 1959 at a slow rate until April 24 when the release rate began to increase as reactor power levels were increased (Pincock 1959). Releases continued throughout the remainder of the test series when the reactor was operating at power levels of 100 kW or greater. The test series involved evaluation of the L2A-1 insert cartridge, which contained both fueled and unfueled BeO ceramic tubes. The objectives of the test were to evaluate the operational effect of water vapor corrosion on fueled BeO tubes and to measure the fission product release rate from uncoated fueled tubes as a function of temperature and operating time.

We calculated an estimated fission product inventory, based on the operational history provided by Pincock (1959), using the RSAC-5 computer code. Through an extensive amount of data analysis related to reported releases, DOE (1991a) estimated release fractions for each run completed as part of IET-14. We elected to use the highest release fractions estimated by DOE (1991a), resulting in assumed release fractions of 2.476×10^{-5}, 0.0025, 0.02476 for solids, halogens, and noble gases, respectively. We assumed the same uranium and ^{41}Ar release values estimated by DOE (1991a).

Pincock (1959) and Boone et al. (1959) report a gross fission product release of 8656 Ci of 10-minute decayed activity. Our reconstructed operation resulted in a total release of more than 49,700 Ci of 10-minute decayed fission products, which is conservative by comparison to the reported release.

DOE (1991a) evaluated IET-14 releases during several separate periods; as such, variable transit times were reported. Because the calculated relative screening value generally increases with increasing transit time, we elected to evaluate the IET-14 releases assuming a relatively slow wind speed and a straight-line trajectory to the nearest offsite location. Consequently, we assumed a wind speed of 4.5 mph (2.0 m s^{-1}), atmospheric stability class D, and a straight-line trajectory to the Building location. These assumptions resulted in the plume passing over Highway 28 (onsite, 11 km downwind) approximately 92 minutes following the release and arriving at the Building (offsite, 14 km downwind) after approximately 117 minutes. Based on these decay times and travel distances, the relative screening value for the IET-14 release for the onsite exposure was 1.3×10^{-7} and for the offsite exposure was 5.4×10^{-6}.

IET-15

The IET-15 test series was performed between May 27 and June 24, 1959. Documented releases of activity began on June 3 and continued through June 24, 1959, when the reactor was operated at power levels exceeding 100 kW (Evans 1959). The test series evaluated the L2C-1 insert cartridge, which was of concentric ring design, with a fuel sheet made of a chromium-UO_2-titanium core clad with an iron-chromium-yttrium alloy. The objectives of the test were to evaluate the endurance capabilities of the advanced metals, the structural and metallurgical integrity of the fuel, the nature and extent of any fuel sheet damage, and the performance potential of the cartridge.

We calculated an estimated fission product inventory, based on the operational history provided by Evans (1959), using the RSAC-5 computer code. Through an extensive amount of data analysis related to reported releases, DOE (1991a) estimated release fractions for each run completed as part of IET-15. We elected to use the highest release fractions estimated by DOE (1991a), resulting in assumed release fractions of 1.02×10^{-6}, 0.000542, and 0.00204 for solids, halogens, and noble gases, respectively. We assumed the same uranium and ^{41}Ar release values estimated by DOE (1991a).

Boone et al. (1959) reports a gross fission product release of 899 Ci of 10-minute decayed activity. Our reconstructed operation resulted in a total release of more than 3400 Ci of 10-minute decayed fission products, which is conservative by comparison to the reported release.

DOE (1991a) evaluated IET-15 releases during several separate periods; as such, variable transit times were reported. Because the calculated relative screening value generally increases with increasing transit time, we elected to evaluate the IET-15 releases assuming a relatively slow wind speed and a straight-line trajectory to the nearest offsite location. Consequently, we assumed a wind speed of 4.5 mph (2.0 m s^{-1}), atmospheric stability class D, and a straight-line trajectory to the Building location. These assumptions resulted in the plume passing over Highway 28 (onsite, 11 km downwind) approximately 92 minutes following the release and arriving at the Building (offsite, 14 km downwind) after approximately 117 minutes. Based on these decay times and travel distances, the relative screening value for the IET-15 release for the onsite exposure was 2.0×10^{-8} and for the offsite exposure was 7.4×10^{-7}.

IET-16

The IET-16 test series was conducted between July 28 and October 9, 1959. This was the first power test of the HTRE-3 reactor assembly, and the objective of the testing was to evaluate operating characteristics of the horizontal core. Releases of radioactivity to the atmosphere were relatively small as a result of this test.

We calculated an estimated fission product inventory, based on a simplified treatment of the operational history provided by Showalter et al. (1959), using the RSAC-5 computer code. Miller et al. (1960) reports that a total of 95 MW-h of power was accumulated during the testing, with a maximum power level of 10 MW. Therefore, we assumed an operating period of 9.5 hours at the maximum power level of 10 MW. DOE (1991a) estimated release fractions based on an extensive review of available data. We determined that these release fractions were appropriate and used the same values, resulting in assumed release fractions of 3.0×10^{-8}, 3.0×10^{-7}, and 6.0×10^{-5} for

solids, halogens, and noble gases, respectively. We assumed the same uranium and ^{41}Ar release values estimated by DOE (1991a).

Showalter et al. (1959) reports a gross fission product release of 1.5 Ci hr^{-1} of 10-minute decayed activity. Assuming this rate of release during the 9.5 hours of assumed testing resulted in a total release of about 15 Ci. Our reconstructed operation resulted in a total release of more than 76 Ci of 10-minute decayed fission products, or a release rate of over 8 Ci hr^{-1} during the assumed 9.5 hours of operation, which is conservative by comparison to the reported release rate.

We assumed a wind speed of 15.2 mph (6.8 m s^{-1}), atmospheric stability class D, and a straight-line trajectory to the nearest offsite location in the general direction of the prevailing winds, based on the transit time and location with the highest dispersion factor reported by DOE (1991a). These assumptions resulted in the plume passing over Highway 20 (onsite, 32 km downwind) approximately 68 minutes following the release and arriving at Cerro Grande (offsite, 45 km downwind) after approximately 96 minutes. Based on these decay times and travel distances, the relative screening value for the IET-16 release for the onsite exposure was 6.4×10^{-11} and for the offsite exposure was 5.0×10^{-11}.

IET-17

The test series designated as IET-17 was performed between October 12 and December 12, 1959. The tests employed the HTRE-2 reactor assembly and assessed the characteristics during power testing of the L2E-1 insert. Releases of radioactivity to the atmosphere were documented to have occurred between November 2 and December 12, 1959, when the reactor was operated at power levels above 100 kW (Pincock 1960a). The test was implemented to evaluate the high temperature characteristics of alumina coated (Al_2O_3) fueled ceramic (BeO) tubes. The L2E-1 insert was a modification of the BeO ceramic insert that was tested previously during the 2B and L2A1 test series. The alumina coating was an attempt to reduce or eliminate the BeO hydrolysis and crystal growths as well as fission product releases. Results indicated the apparent effectiveness of the coating in reducing the evolution of some fission products (Evans 1960).

We calculated an estimated fission product inventory, based on the operational history provided by Pincock (1960a), using the RSAC-5 computer code. Through an extensive amount of data analysis related to reported releases, DOE (1991a) estimated release fractions for each run completed as part of IET-17. We initially selected the highest release fractions estimated by DOE (1991a), but this resulted in a fission product release rate that was not conservative compared to release rates reported by Pincock (1960a). Therefore, we increased these release fractions by a factor of 10, resulting in assumed release fractions of 2.73×10^{-6}, 4.75×10^{-5}, 5.45×10^{-3} for solids, halogens, and noble gases, respectively. We assumed the same uranium and ^{41}Ar release values estimated by DOE (1991a).

Pincock (1960a) reports a gross fission product release of 2017 Ci of 10-minute decayed activity. Our reconstructed operation resulted in a total release of more than 5600 Ci of 10-minute decayed fission products, which is conservative by comparison to the reported release.

DOE (1991a) evaluated IET-17 releases during several separate periods; as such, variable transit times were reported. Because the calculated relative screening value generally increases with increasing transit time, we elected to evaluate the IET-17 releases assuming a relatively slow wind speed and a straight-line trajectory to the nearest offsite location. Consequently, we assumed a wind speed of 4.5 mph (2.0 m s^{-1}), atmospheric stability class D, and a straight-line

trajectory to the Building location. These assumptions resulted in the plume passing over Highway 28 (onsite, 11 km downwind) approximately 92 minutes following the release and arriving at the Building (offsite, 14 km downwind) after approximately 117 minutes. Based on these decay times and travel distances, the relative screening value for the IET-17 release for the onsite exposure was 1.6×10^{-8} and for the offsite exposure was 9.4×10^{-8}.

IET-18

The IET-18 test series was designated as the Phase 2 testing of the HTRE-3 reactor assembly and was conducted between December 23, 1959 and February 8, 1960. This test series was an extension of the IET-16 test series, designed to help define reactor operational parameters.

We calculated an estimated fission product inventory, based on the operational history provided by Highberg et al. (1960) and further refined by DOE (1991a), using the RSAC-5 computer code. Through an extensive amount of data analysis related to reported releases, DOE (1991a) estimated release fractions for each major period of reactor operation completed as part of IET-18. We initially used the highest release fractions estimated by DOE (1991a); however, this did not result in a conservative release by comparison to the reported gross fission product release. Therefore, we increased the halogen release fraction by a factor of 10 and the solid and noble gas release fractions by a factor of 100, resulting in assumed release fractions of 1.28×10^{-6}, 5.52×10^{-4}, and 2.59×10^{-3} for solids, halogens, and noble gases, respectively. We assumed the same uranium and ^{41}Ar release values estimated by DOE (1991a).

Highberg et al. (1960) reports a gross fission product release of 1157 Ci of 10-minute decayed activity. Our reconstructed operation resulted in a total release of more than 18,600 Ci of 10-minute decayed fission products, which is conservative by comparison to the reported release.

DOE (1991a) evaluated IET-18 releases during several separate periods; as such, variable transit times were reported. Because the calculated relative screening value generally increases with increasing transit time, we elected to evaluate the IET-18 releases assuming a relatively slow wind speed and a straight-line trajectory to the nearest offsite location. Consequently, we assumed a wind speed of 4.5 mph (2.0 m s^{-1}), atmospheric stability class D, and a straight-line trajectory to the Building location. These assumptions resulted in the plume passing over Highway 28 (onsite, 11 km downwind) approximately 92 minutes following the release and arriving at the Building (offsite, 14 km downwind) after approximately 117 minutes. Based on these decay times and travel distances, the relative screening value for the IET-18 release for the onsite exposure was 1.2×10^{-7} and for the offsite exposure was 5.4×10^{-6}.

IET-19

The IET-19 test series was conducted between February 9 and April 30, 1960. Releases of radioactivity to the atmosphere were documented to have occurred between February 17 and April 30, 1960, when the reactor was operated at power levels above 100 kW. The test series involved evaluation of the L2E-3 insert cartridge, which contained both fueled and unfueled hexagonal BeO ceramic tubes coated on the inside surface with zirconia (ZrO_2). The primary objectives of the test series were to evaluate the effectiveness of the zirconia coating against hydrolysis and the release of fission products, to determine fission product release as a function of

insert temperature, and to obtain information regarding the effectiveness of an electrostatic precipitator in removing fission products from the reactor effluent (Pincock 1960b).

We calculated an estimated fission product inventory, based on the operational history provided by Pincock (1960b), using the RSAC-5 computer code. Through an extensive amount of data analysis related to reported releases, DOE (1991a) estimated release fractions for each run completed as part of IET-19. We elected to use the highest release fractions estimated by DOE (1991a), resulting in assumed release fractions of 3.99×10^{-6}, 0.0013, and 0.00798 for solids, halogens, and noble gases, respectively. We assumed the same uranium and ^{41}Ar release values estimated by DOE (1991a).

Pincock (1960b) reports a gross fission product release of 2908 Ci of 10-minute decayed activity. Our reconstructed operation resulted in a total release of more than 14,000 Ci of 10-minute decayed fission products, which is conservative by comparison to the reported release.

DOE (1991a) evaluated IET-19 releases during several separate periods; as such, variable transit times were reported. Because the calculated relative screening value generally increases with increasing transit time, we elected to evaluate the IET-19 releases assuming a relatively slow wind speed and a straight-line trajectory to the nearest offsite location. Consequently, we assumed a wind speed of 4.5 mph (2.0 m s^{-1}), atmospheric stability class D, and a straight-line trajectory to the Building location. These assumptions resulted in the plume passing over Highway 28 (onsite, 11 km downwind) approximately 92 minutes following the release and arriving at the Building (offsite, 14 km downwind) after approximately 117 minutes. Based on these decay times and travel distances, the relative screening value for the IET-19 release for the onsite exposure was 6.6×10^{-8} and for the offsite exposure was 3.1×10^{-6}.

IET-20

The IET-20 test series was conducted between May 11 and June 13, 1960. Releases of radioactivity to the atmosphere were documented to have occurred between May 14 and June 10, 1960, when the reactor was operated at power levels above 100 kW. The test series involved evaluation of the L2E-2 insert cartridge, which contained both fueled and unfueled hexagonal BeO ceramic tubes coated on the inside surface with alumina (Al_2O_3). The primary objectives of the test series were to operate the insert cartridge at a peak temperature of 2500° F for 25 hours and then at a peak temperature of 2600° F for 100 hours; to evaluate the performance of the assembly with respect to aerothermodynamics, structural integrity, fission product release, and hydrolysis; and to obtain information about the effectiveness of a precipitator in removing fission products from the reactor effluent (Foster et al. 1960).

We calculated an estimated fission product inventory, based on the operational history provided by Foster et al. (1960), using the RSAC-5 computer code. Through an extensive amount of data analysis related to reported releases, DOE (1991a) estimated release fractions for each run completed as part of IET-20. We elected to use the highest release fractions estimated by DOE (1991a), resulting in assumed release fractions of 1.02×10^{-5}, 0.000862, and 0.0204 for solids, halogens, and noble gases, respectively. We assumed the same uranium and ^{41}Ar release values estimated by DOE (1991a).

Foster et al. (1960) reports a gross fission product release of 5119 Ci of 10-minute decayed activity. Our reconstructed operation resulted in a total release of more than 26,000 Ci of 10-minute decayed fission products, which is conservative by comparison to the reported release.

DOE (1991a) evaluated IET-20 releases during several separate periods; as such, variable transit times were reported. Because the calculated relative screening value generally increases with increasing transit time, we elected to evaluate the IET-20 releases assuming a relatively slow wind speed and a straight-line trajectory to the nearest offsite location. Consequently, we assumed a wind speed of 4.5 mph (2.0 m s^{-1}), atmospheric stability class D, and a straight-line trajectory to the Building location. These assumptions resulted in the plume passing over Highway 28 (onsite, 11 km downwind) approximately 92 minutes following the release and arriving at the Building (offsite, 14 km downwind) after approximately 117 minutes. Based on these decay times and travel distances, the relative screening value for the IET-20 release for the onsite exposure was 5.2×10^{-8} and for the offsite exposure was 1.7×10^{-6}.

IET-21

The IET-21 test series, or Fuel Element Effluent Test 1 (FEET 1), was conducted between June 20 and August 8, 1960. Releases of radioactivity to the atmosphere were documented to have occurred between June 29 and August 6, 1960, when the reactor was operated at power levels above 100 kW. The test series involved evaluation of the L2A-2 insert cartridge, which consisted of uncoated fueled and unfueled BeO tubes. The primary objectives of the test series were to provide a source suitable for evaluating the effectiveness of the electrostatic precipitator, obtain further information pertaining to the release of fission products as a function of temperature from uncoated BeO fueled tubes, and study atmospheric diffusion of fission products under various meteorological conditions (Pincock 1960c).

We calculated an estimated fission product inventory, based on the operational history provided by Pincock (1960c), using the RSAC-5 computer code. Through an extensive amount of data analysis related to reported releases, DOE (1991a) estimated release fractions for each run completed as part of IET-21. We elected to use the highest release fractions estimated by DOE (1991a), resulting in assumed release fractions of 1.76×10^{-5}, 0.000784, and 0.0176 for solids, halogens, and noble gases, respectively. We assumed the same uranium and ^{41}Ar release values estimated by DOE (1991a).

Pincock (1960c) reports a gross fission product release of 2688 Ci of 10-minute decayed activity. Our reconstructed operation resulted in a total release of more than 4700 Ci of 10-minute decayed fission products, which is conservative by comparison to the reported release.

DOE (1991a) evaluated IET-21 releases during several separate periods; as such, variable transit times were reported. Because the calculated relative screening value generally increases with increasing transit time, we elected to evaluate the IET-21 releases assuming a relatively slow wind speed and a straight-line trajectory to the nearest offsite location. Consequently, we assumed a wind speed of 4.5 mph (2.0 m s^{-1}), atmospheric stability class D, and a straight-line trajectory to the Building location. These assumptions resulted in the plume passing over Highway 28 (onsite, 11 km downwind) approximately 92 minutes following the release and arriving at the Building (offsite, 14 km downwind) after approximately 117 minutes. Based on these decay times and travel distances, the relative screening value for the IET-21 release for the onsite exposure was 1.0×10^{-8} and for the offsite exposure was 2.5×10^{-7}.

IET-22

The IET-22 test series, or Limited Melt Experiment No. 1 (LIME-I), was conducted between August 12 and 25, 1960. The test series involved evaluation of the L2E-4 insert cartridge, which consisted of uncoated fueled and unfueled BeO tubes. The purpose of the test series was to operate the ceramic cartridge for 10 minutes at a power level sufficient to cause portions of the plugged fuel region to melt, to evaluate the nature and propagation of such a melt, and to verify the ability to predict such phenomena (Pincock 1960d).

We calculated an estimated fission product inventory, based on the operational history provided by Pincock (1960d), using the RSAC-5 computer code. Through an extensive amount of data analysis related to reported releases, DOE (1991a) estimated release fractions for each run completed as part of IET-22. We elected to use the highest release fractions estimated by DOE (1991a), resulting in assumed release fractions of 4.52×10^{-3}, 0.125, and 1.0 for solids, halogens, and noble gases, respectively. We assumed the same uranium and ^{41}Ar release values estimated by DOE (1991a).

Pincock (1960d) reports a gross fission product release of 3250 Ci of 10-minute decayed activity. Our reconstructed operation resulted in a total release of more than 405,000 Ci of 10-minute decayed fission products, which is very conservative by comparison to the reported release.

We assumed a wind speed of 21.1 mph (9.4 m s^{-1}), atmospheric stability class D, and a straight-line trajectory to the nearest offsite location in the general direction of the prevailing winds, based on the transit time and location with the highest dispersion factor reported by DOE (1991a). These assumptions resulted in the plume passing over Highway 28 (onsite, 10 km downwind) approximately 18 minutes following the release and arriving at Monteview (offsite, 17 km downwind) after approximately 30 minutes. Based on these decay times and travel distances, the relative screening value for the IET-22 release for the onsite exposure was 1.8×10^{-7} and for the offsite exposure was 5.0×10^{-7}.

IET-23

The IET-23 test series, or Fuel Element Effluent Test 2 (FEET 2), was conducted between September 1 and October 14, 1960. Releases of radioactivity to the atmosphere were documented to have occurred between September 7 and October 14, 1960, when the reactor was operated at power levels above 100 kW. The test series involved the continued evaluation of the L2A-2 insert cartridge, used for IET-21 (Pincock 1960c).

We calculated an estimated fission product inventory, based on the operational history provided by Pincock (1960c), using the RSAC-5 computer code. Through an extensive amount of data analysis related to reported releases, DOE (1991a) estimated release fractions for each run completed as part of IET-23. We elected to use the highest release fractions estimated by DOE (1991a), resulting in assumed release fractions of 1.84×10^{-5}, 0.00168, and 0.0184 for solids, halogens, and noble gases, respectively. We assumed the same uranium and ^{41}Ar release values estimated by DOE (1991a).

Pincock (1960c) reports a gross fission product release of 2125 Ci of 10-minute decayed activity. Our reconstructed operation resulted in a total release of more than 5500 Ci of 10-minute decayed fission products, which is conservative by comparison to the reported release.

DOE (1991a) evaluated IET-23 releases during several separate periods; as such, variable transit times were reported. Because the calculated relative screening value generally increases with increasing transit time, we elected to evaluate the IET-23 releases assuming a relatively slow wind speed and a straight-line trajectory to the nearest offsite location. Consequently, we assumed a wind speed of 4.5 mph (2.0 m s^{-1}), atmospheric stability class D, and a straight-line trajectory to the Building location. These assumptions resulted in the plume passing over Highway 28 (onsite, 11 km downwind) approximately 92 minutes following the release and arriving at the Building (offsite, 14 km downwind) after approximately 117 minutes. Based on these decay times and travel distances, the relative screening value for the IET-23 release for the onsite exposure was 1.2×10^{-8} and for the offsite exposure was 4.4×10^{-7}.

IET-24

The IET-24 test series, also referred to as LIME-II or Sub-LIME, was conducted between October 17 and 26, 1960. The test series involved evaluation of the L2E-5 insert cartridge, which consisted of uncoated fueled and unfueled BeO tubes and was designed to simulate a condition where air flow was restricted to 10% of the normal flow. This is in contrast to LIME-I, which was designed to ensure melting by completely blocking off coolant air (Pincock 1960e; Baker 1961).

We calculated an estimated fission product inventory, based on the operational history provided by Pincock (1960e), using the RSAC-5 computer code. Through an extensive amount of data analysis related to reported releases, DOE (1991a) estimated release fractions for each run completed as part of IET-24. We elected to use the release fractions estimated by DOE (1991a), resulting in assumed release fractions of 8.2×10^{-5}, 0.03, and 0.328 for solids, halogens, and noble gases, respectively. We assumed the same uranium and ^{41}Ar release values estimated by DOE (1991a).

Pincock (1960e) reports a gross fission product release of 1880 Ci of 10-minute decayed activity. Our reconstructed operation resulted in a total release of more than 340,000 Ci of 10-minute decayed fission products, which is very conservative by comparison to the reported release.

We assumed a wind speed of 21.7 mph (9.7 m s^{-1}), atmospheric stability class D, and a straight-line trajectory to the nearest offsite location in the general direction of the prevailing winds, based on the transit time and location with the highest dispersion factor reported by DOE (1991a). These assumptions resulted in the plume passing over Highway 28 (onsite, 11 km downwind) approximately 19 minutes following the release and arriving at the Building (offsite, 14 km downwind) after approximately 24 minutes. Based on these decay times and travel distances, the relative screening value for the IET-24 release for the onsite exposure was 1.6×10^{-7} and for the offsite exposure was 4.3×10^{-7}.

IET-25

The IET-25 test series was an extension of the Phase 2 testing of the HTRE-3 reactor assembly and was conducted between November 15 and December 16, 1960. This test series was an extension of the IET-18 test series, and it was designed to demonstrate the capabilities of the fuel elements above design temperatures and to confirm that the power plant could achieve a full nuclear start as predicted (Linn et al. 1962, cited in DOE 1991a).

We calculated an estimated fission product inventory, based on the operational history provided by Highberg et al. (1961) and further refined by DOE (1991a), using the RSAC-5 computer code. Through an extensive amount of data analysis related to reported releases, DOE (1991a) estimated release fractions for each major period of reactor operation completed as part of IET-25. We initially used the highest release fractions estimated by DOE (1991a); however, this did not result in a conservative release compared to the reported gross fission product release. Therefore, we increased all release fractions by a factor of 10, resulting in assumed release fractions of 7.45×10^{-8}, 1.49×10^{-4}, and 1.49×10^{-4} for solids, halogens, and noble gases, respectively. We assumed the same uranium and ^{41}Ar release values estimated by DOE (1991a).

Highberg et al. (1961) reports a gross fission product release of 218 Ci of 10-minute decayed activity. Our reconstructed operation resulted in a total release of more than 1740 Ci of 10-minute decayed fission products, which is conservative by comparison to the reported release.

DOE (1991a) evaluated IET-25 releases during several separate periods; as such, variable transit times were reported. Because the calculated relative screening value generally increases with increasing transit time, we elected to evaluate the IET-25 releases assuming a relatively slow wind speed and a straight-line trajectory to the nearest offsite location. Consequently, we assumed a wind speed of 4.5 mph (2.0 m s^{-1}), atmospheric stability class D, and a straight-line trajectory to the Building location. These assumptions resulted in the plume passing over Highway 28 (onsite, 11 km downwind) approximately 92 minutes following the release and arriving at the Building (offsite, 14 km downwind) after approximately 117 minutes. Based on these decay times and travel distances, the relative screening value for the IET-25 release for the onsite exposure was 3.9×10^{-8} and for the offsite exposure was 8.2×10^{-7}.

IET-26

The IET-26 test series was conducted between December 22, 1960 and March 31, 1961. Releases of radioactivity to the atmosphere were assumed to have occurred between December 23, 1960 and March 30, 1961, when the reactor was operated at power levels above 130 kW. The test series involved the continued evaluation of the L2E-6 insert cartridge, which consisted of fueled and unfueled BeO hexagonal tubes coated on the inner surface with ZrO_2. The objectives of the test series were to evaluate the ZrO_2 coating at temperatures above the design conditions and to operate the insert at a fuel temperature ranging from 2500 to 2800°F to better understand fission product release behavior as a function of time and temperature (Field et al. 1961).

We calculated an estimated fission product inventory, based on the operational history provided by Field et al. (1961), using the RSAC-5 computer code. Through an extensive amount of data analysis related to reported releases, DOE (1991a) estimated release fractions for each run completed as part of IET-26. We initially used the highest release fractions estimated by DOE (1991a); however, this did not result in a conservative release by comparison to the reported gross fission product release. Therefore, we elected to increase all release fractions by a factor of 10, resulting in assumed release fractions of 1.94×10^{-5}, 0.00359, and 0.0387 for solids, halogens, and noble gases, respectively. We assumed the same uranium and ^{41}Ar release values estimated by DOE (1991a).

Field et al. (1961) reports a gross fission product release rate of between 4 and 18 Ci hr^{-1}. Conservatively assuming an 18 Ci hr^{-1} release rate during the entire 399 hours of operation above 130 kW resulted in a total particulate release of 7,180 Ci. Our reconstructed operation resulted in

a total release of more than 46,400 Ci of 10-minute decayed fission products, or a release rate of over 116 Ci hr^{-1} during the assumed 399 hours of operation, which is conservative by comparison to the reported release rate.

DOE (1991a) evaluated IET-26 releases during several separate periods; as such, variable transit times were reported. Because the calculated relative screening value generally increases with increasing transit time, we elected to evaluate the IET-26 releases assuming a relatively slow wind speed and a straight-line trajectory to the nearest offsite location. Consequently, we assumed a wind speed of 4.5 mph (2.0 m s^{-1}), atmospheric stability class D, and a straight-line trajectory to the Building location. These assumptions resulted in the plume passing over Highway 28 (onsite, 11 km downwind) approximately 92 minutes following the release and arriving at the Building (offsite, 14 km downwind) after approximately 117 minutes. Based on these decay times and travel distances, the relative screening value for the IET-26 release for the onsite exposure was 1.2×10^{-7} and for the offsite exposure was 4.3×10^{-6}.

Fuel Element Burn Tests

The Fuel Element Burn Tests (FEBTs) A and B were conducted on test GRID III at the INEEL in support of the General Electric Aircraft Nuclear Propulsion Department Program to evaluate the consequences of a nuclear aircraft crash involving a fire. Aged fuel elements were heated to assist with understanding the behavior of a fuel element engrossed in a large fire and to provide some initial data on the percentage release of fission products to the environment (Brodsky and Beard 1960). Meteorological conditions had been carefully studied in advance and were closely monitored during the tests to allow for the collection of as much experimental data as possible. Because these tests employed aged-fuel elements and the transit time to the site boundary represents a relatively insignificant additional time for decay, evaluating exposure at different decay times is unnecessary.

FEBT-A

A pool of jet fuel was ignited under the FEBT-A fuel element, which contained approximately 5000 Ci of fission products, at 2:19 p.m. MST on March 20, 1957. The fuel element burned for about 2 hours and reached a temperature of approximately 2250°F. Following the fire, the fuel element was intact with a small puncture in the cladding. The 15-minute average wind directions at 250 feet from 2:15 p.m. through 4:00 p.m. ranged from 200 to 218° (coming from the southwest), and the average wind velocities were 8 to 12 mph. The 20-foot altitude winds ranged from 200 to 210° at 13 to 14 mph. The vertical temperature stratification during the burning period showed temperature decreases of 3 to 4°F from the ground to the 250-foot level, indicating lapse (i.e., increasing temperature with increasing elevation, which allows for greater upward dispersion), and, therefore, optimum diffusion conditions (Brodsky and Beard 1960).

Brodsky and Beard (1960) reports the fission product composition for this fuel element and noted that the inventory was "based on 20 MW elements at 120 hour operation plus 70 days immediate decay". DOE (1991a) attempted to reconstruct the fission product inventory and noted that this operating history was not sufficient to calculate an inventory without making some additional assumptions. In an iterative process, the authors adjusted the power level for the RSAC-4 computer code (Wenzel 1990) calculations until the calculated ^{137}Cs and ^{144}Ce

inventories were within 10% of the values reported by Brodsky and Beard (1960). We made a similar assumption and set the reactor power level to 0.56 MW for 120 hours followed by a decay of 70 days for our inventory calculation with the RSAC-5 computer code. Uranium releases were estimated by assuming a total 90% enriched uranium content of 222 g for each element.

Although the FEBT experiments were designed to provide initial data on the release of fission products to the environment, Brodsky and Beard (1960) reports only that "probably less than a few tenths of a per cent" of the fuel element inventory was released. Therefore, we have elected to use the release fractions that were used for the FPFRT runs that were conducted at the anticipated operating temperatures (see Tables 15 and 16). Although the temperature reached during FEBT-A was somewhat less than that reached during the FPFRT runs (2250°F compared to 2912°F), the element was heated for a longer period of time, so similar release fractions were considered justified. For all other radionuclides not measured during the FPFRT releases, release fractions of 0.1, 0.5, and 1.0 were used for solids (including uranium in the fuel element), halogens, and noble gases, respectively.

We assumed a wind speed of 13 mph (5.8 m s^{-1}), atmospheric stability D, and a straight-line trajectory to the nearest offsite location in the general direction of the prevailing winds, considering the location with the highest dispersion factor reported by DOE (1991a). These assumptions resulted in the plume passing over Highway 33 (onsite, 25 km downwind) and Reno Ranch (offsite, 48 km downwind). DOE (1991a) reported a transit time to the INEEL site boundary of 1.9 hours (114 minutes), which corresponded to a wind speed of 7.0 m s^{-1}. Therefore, we also evaluated exposure to the plume in Reno Ranch, assuming a wind speed of 7.0 m s^{-1}. Based on these wind speeds and travel distances, the relative screening value for the FEBT-A release for the onsite exposure was 2.6×10^{-9} and for the offsite exposure was 9.6×10^{-8} for the 5.8 m s^{-1} wind speed and 8.0×10^{-8} for the 7.0 m s^{-1} wind speed.

FEBT-B

The second test, FEBT-B, began at 6:47 p.m. MST on March 20, 1957. A fuel element containing approximately 10,000 Ci of fission products was heated in a furnace by an oxygen-fed fire of thermite, steel wool, and iron filings. The fuel element was heated to approximately 5000°F and continued burning for about 4 minutes. Most of the fuel element was melted and dispersed within 90 seconds. Wind speeds measured 20 feet above the ground at the test site for the first two 15-minute periods averaged 7 mph from 215° and 6 mph from 210°, with a total variation from 195 to 235°. The vertical temperature variation had changed to an inversion condition, with temperature increases of 1 to 2°F from the ground to the 250-foot level. This inversion prevented the cloud from rising to higher levels and gave poor diffusion conditions (Brodsky and Beard 1960).

Brodsky and Beard (1960) reports only the relative yield for the fission product composition of the FEBT-B fuel element and states that it was based on 6.16×10^{21} fissions and immediate 250-day cooling. Again, DOE (1991a) attempted to reconstruct the fission product inventory and noted that this operating history was not sufficient to calculate an inventory without making some additional assumptions. Using the conversion factor of 3.12×10^{10} fissions per W-s and assuming the same type of fuel element operated in the same reactor as for FEBT-A, DOE (1991a) concluded that a reactor operation of 4.022 days at a power level of 0.568 MW would be necessary to produce the burnup implied by the number of fissions reported by Brodsky and

Beard (1960). However, these operating parameters lead to a fission product inventory less than that for FEBT-A and considerably less than the 10,000 Ci reported inventory; therefore the authors adjusted the reactor operating period to achieve an inventory of approximately twice that for the FEBT-A fuel element. We elected to use a similar approach to estimate the FEBT-B fuel element inventory and arrived at a calculated inventory of approximately 10,000 Ci. We set the reactor power level to 0.56 MW for 69 days followed by a decay of 250 days for our inventory calculation with the RSAC-5 computer code. Uranium releases were estimated by assuming a total 90% enriched uranium content of 222 g for each element.

Again, Brodsky and Beard (1960) reported only that "the fractional activity released was estimated from field measurements and filter paper analyses to be a maximum of 10 per cent, probably much less." Because this does not provide us with specific release fractions for individual radionuclides, we elected to select the same release fractions used for FPFRT-8 (Table 18). Although the FPFRT-8 run resulted in a lower temperature than that achieved during FEBT-B (4172°F compared to 5000°F), the fuel element was maintained at this temperature for a shorter period of time during FEBT-B (4 minutes compared to 10 minutes) so similar release fractions were considered justified for this screening analysis.

We assumed a wind speed of 6 mph (2.7 m s^{-1}), atmospheric stability class F, and a straight-line trajectory to the nearest offsite location in the general direction of the prevailing winds, considering the location with the highest dispersion factor reported by DOE (1991a). These assumptions resulted in the plume passing over Highway 33 (onsite, 20 km downwind) and Birch Creek (offsite, 57 km downwind). DOE (1991a) reported a transit time to the INEEL site boundary of 2.8 hours (168 minutes), which corresponded to a wind speed of 5.7 m s^{-1}. Therefore, we also evaluated exposure to the plume in Birch Creek, assuming a wind speed of 5.7 m s^{-1}. Based on these wind speeds and travel distances, the relative screening value for the FEBT-B release for the onsite exposure was 2.2×10^{-7} and for the offsite exposure was 1.8×10^{-5} for the 2.7 m s^{-1} wind speed and 8.3×10^{-6} for the 5.7 m s^{-1} wind speed.

Fission Products Field Release Tests

The Air Research and Development Command of the U.S. Air Force sponsored a series of tests at the INEEL from July 25 through September 26, 1958 to obtain information about the release of radioactivity from potential accidents involving nuclear powered aircraft. A total of nine FPFRTs were conducted in an effort to evaluate release percentages, airborne radioactivity, and diffusion and deposition characteristics of fission products released from melted aircraft reactor fuel elements (Convair 1959). Because these tests employed aged-fuel elements and the transit time to the site boundary represented a relatively insignificant additional time for decay, evaluating exposure at different decay times was unnecessary.

Instruments situated about a fan-shaped grid with seven concentric arcs and a maximum radius of about 5 miles were used to obtain cloud diffusion, meteorological, radiological, radiobiological, and deposition data. Release percentages, aerosol sizes, deposition velocities, external and internal doses, fluorescent tracer behavior, and atmospheric diffusion parameters were determined during the tests (Convair 1959). The tests were conducted under strict operational controls and very carefully monitored meteorological conditions, which are described in detail for each release by Convair (1959) and Wehman (1959). Specific conditions existing during each test are compiled in Table 15.

Of the nine separate tests, five were conducted with "aged" (decayed for 922 to 985 days) fuel elements, and four were conducted with "fresh" elements (decayed for 42 to 65 days) (Wehman 1959). The operational histories and estimated fission product inventories for each of the elements are provided by Convair (1959). To simulate a potential accident, an induction type furnace was used to rapidly heat the elements to the melting point in approximately 2 minutes and maintain this temperature for approximately 10 minutes after melting began. Convair (1959) discusses operating details related to the furnace that was used for the tests.

Table 15. Meteorological Conditions during the Fission Products Field Release Tests[a]

Test Number (date and time)	METEOROLOGICAL PARAMETER	Time after release (min)				
		0	15	30	45	60
1 (7/25/58 at 6:09 p.m.)	Average wind speed (m s^{-1})[b]	5.5	3.6	4.6		
	Prevailing wind direction[b, c]	238°	236°	271°		
	Dispersion conditions[d]	+0.6	-1.3	-2.2		
2 (8/4/58 at 8:16 p.m.)	Average wind speed (m s^{-1})	3.4	4.2	4.5		
	Prevailing wind direction	265°	260°	261°		
	Dispersion conditions	-7.5	-8.1	-9.0		
3 (8/6/58 at 7:15 p.m.)	Average wind speed (m s^{-1})	2.9	3.2	2.0		
	Prevailing wind direction	233°	230°	234°		
	Dispersion conditions	-2.4	-3.0	-8.0	-10.0	msg[e]
4 (8/14/58 at 6:18 p.m.)	Average wind speed (m s^{-1})	5.9	4.9	5.4	4.4	4.8
	Prevailing wind direction	236°	248°	220°	238°	244°
	Dispersion conditions	+0.8	+1.6	+1.2	msg	msg
5 (8/27/58 at 5:20 p.m.)	Average wind speed (m s^{-1})	7.3	8.7	8.5	7.2	
	Prevailing wind direction	215°	233°	245°	247°	
	Dispersion conditions	+2.0	+1.2	+0.8	+1.0	
6 (9/4/58 at 5:12 p.m.)	Average wind speed (m s^{-1})	6.1	6.3	7.7	msg	msg
	Prevailing wind direction	223°	231°	226°	msg	msg
	Dispersion conditions	+0.4	+1.2	+0.8	msg	msg
7 (9/17/58 at 7:04 p.m.)	Average wind speed (m s^{-1})	5.6	4.9	msg	6.4	5.7
	Prevailing wind direction	209°	217°	215°	227°	227°
	Dispersion conditions	-2.2	-1.8	-3.6	msg	msg
8 (9/18/58 at 6:31 p.m.)	Average wind speed (m s^{-1})	4.5	5.0	4.7	4.9	4.0
	Prevailing wind direction	211°	209°	213°	217°	220°
	Dispersion conditions	-2.8	-3.8	-3.8	-3.8	-4.0
9 (9/26/58 at 4:21 p.m.)	Average wind speed (m s^{-1})	1.1	1.8	2.0	3.0	3.8
	Prevailing wind direction	227°	193°	197°	227°	215°
	Dispersion conditions	+0.8	-0.8	+0.0	-1.6	+0.0

[a] Source: Wehman (1959).

[b] Measured at a height of 3 meters.

[c] Direction from which the wind was blowing.

[d] Reported as °C per 100 m rise in elevation, negative and positive values indicate inversion and lapse conditions, respectively.

[e] Reported by Wehman (1959), presumed to mean "missing."

Based on the reactor operating parameters provided by Convair (1959), we used the RSAC-5 computer code to calculate the expected fission product inventory for each of the nine fuel elements. This calculated inventory was then compared to the inventory reported by Convair (1959), and the highest value for a given radionuclide was selected for screening purposes. Table 16 lists the calculated and reported quantities for several radionuclides present in the fuel element used during FPFRT-5. The two inventories were generally consistent, with the most notable exception being the approximately factor of ten lower quantities of ^{127m}Te and ^{129m}Te calculated using RSAC-5. Comparing the calculated and reported inventories of the remaining eight fuel elements yielded the same general consistency between the two inventories, with the calculated amounts of ^{127m}Te and ^{129m}Te approximately a factor of 10 lower. For the purpose of being conservative for this screening analysis, we used the higher of the two values for our evaluation of each test. Uranium releases were estimated by assuming a total 90% enriched uranium content of 222 g for each element.

Table 16. Comparison of Calculated and Reported Radionuclide Inventories for the Fuel Element Used During FPFRT-5

Nuclide	Calculated inventory[a] (Ci)	Reported inventory[b] (Ci)	Highest value
Sr- 89	51.3	54	54
Sr- 90	0.538	0.62	0.62
Y- 91	59.3	63	63
Zr- 95	62	64	64
Nb- 95	45.8	79	79
Ru-103	36.2	35	36.2
Ru-106	1.05	1	1.05
Te-127m	0.193	1.8	1.8
Te-129m	1.2	12	12
I-131	7.39	7.8	7.8
Cs-137	0.573	0.5	0.573
Ba-140	42	45	45
La-140	48.3	45	48.3
Ce-141	69.6	74	74
Ce-144	17.2	21	21
Pr-143	48	48	48
Pr-144	17.2	21	21
Nd-147	11.9	15	15
Pm-147	2.21	2.8	2.8

[a] Calculated using RSAC-5 computer code
[b] Source: Convair (1959)

To determine the amount of material that may have actually been emitted to the atmosphere during these tests, it was necessary to estimate release fractions for the radionuclides present in the fuel elements. Melting a fuel element certainly resulted in the release of some portion of its contents, but the relative amount of material that is released typically varies considerably by radionuclide and was not a constant for the fuel element as a whole. It is a somewhat difficult task

to estimate these release fractions, and conservative estimates of radionuclides by group are often used (e.g., 0.1 for solids, 0.5 for halogens, and 1.0 for noble gases, as was done for the radionuclide selection calculations). However, one of the objectives of the FPFRT Program was to estimate release fractions during the tests, and these measured release fractions were used to the extent possible for the screening calculations. Several methods were employed for this purpose during the tests and are described in detail by Convair (1959).

The first method used to estimate screening factors involved pre- and post-melt gamma spectrum analysis. This method was used for the four "aged" element tests, and ^{137}Cs was the only isotope escaping in sufficient quantity to determine release percentages using this method.

The second method involved measuring the pre- and post-melt gamma dose rate levels using an ionization chamber with a fixed geometry. This technique provided values that were generally consistent with the gamma spectrum analysis method, but the dose rate levels yielded slightly lower values because they included the contribution of gamma-emitting radionuclides other than ^{137}Cs.

Air samples collected on pleated fiberglass filters at the 100-meter arc of the sampling network provided a third means of estimating release fractions. This method involved the use of the release fraction measured for ^{137}Cs by pre- and post-melt gamma spectroscopy or an estimated release fraction for ^{131}I. For example, by using the ^{137}Cs release fraction, the ^{137}Cs inventory, the ^{137}Cs collected on the filter, the ^{90}Sr inventory, and the ^{90}Sr collected on the filter, the ^{90}Sr release fraction can be estimated. The validity of this procedure, however, depends on the assumption that all of the released cesium is in aerosol form, that it is filtered with the same efficiency as the strontium, and that it is released at the same rate as the strontium.

A fourth technique for estimating release fractions involved the use of fractional sampling of the effluent at the furnace. The contents of 11 sequential evacuated bottles and an integrating bottle were analyzed to determine the release fractions for several nuclides as well as the relative time of their release.

Finally, maximum or upper bound release fractions can be inferred from pre- and post-melt radiochemical analyses of some of the fuel element specimens and their post-release residues. However, because of difficulties experienced in dissolving some of the residue, these analyses can only be used to obtain upper limits for the isotopic release fractions.

For the purpose of this screening analysis, we selected the highest, or most conservative, measured release fractions. However, the maximum temperatures attained during the tests were not constant because furnace malfunction during FPFRT-4 and FPFRT-8 resulted in higher than anticipated temperatures. Table 17 shows the maximum attained temperatures for each test, based on temperature measurements with the furnace at thermal equilibrium in the range of 1000 to 1600°C (Convair 1959). The upper end of this range is assumed to be the anticipated operating temperature of the furnace for FPFRT-1, -2, -3, -5, -6, -7, and -9. Convair (1959) provides estimates of the increase above the anticipated operating temperature that occurred during FPFRT-4 (300 to 400°C) and FPFRT-8 (500 to 700°C). The upper estimate of these values was added to the anticipated operating temperatures for these two tests (i.e., 700°C is added to 1600°C for FRFRT-8).

The higher temperatures likely resulted in larger release fractions, but this was not indicated by all of the release fraction measurements. For example, the release fractions as determined by network air samples are higher for FPFRT-1 than for FPFRT-4 for ^{90}Sr and are higher for FPFRT-7 than for FPFRT-8 for $^{95}Zr/Nb$. However, the release fractions as measured by pre- and

post-melt gamma spectrum analysis and dose rate measurements do suggest a nearly factor of 2 higher release fraction for ^{137}Cs and gamma-emitting radionuclides for FPFRT-4 than for FPFRT-1, -2, or -3.

Table 17. Approximate Furnace Temperatures Attained during the FPFRT Program[a]

Test #	1	2	3	4	5	6	7	8	9
°C	1600	1600	1600	2000	1600	1600	1600	2300	1600
°F	2912	2912	2912	3632	2912	2912	2912	4172	2912

[a] Source: Convair (1959)

For this screening analysis, we used the release fractions for ^{137}Cs as measured by pre- and post-melt gamma spectrum analysis. This was determined for FPFRT-1, -2, -3, and -4. The highest measured value for the first three tests at anticipated operating temperatures was 0.51, and the value measured for FPFRT-4 was 0.83, an increase of approximately 63%. The upper bound value for a ^{137}Cs release fraction as measured by pre- and post-melt radiochemical analysis was 0.97 for FPFRT-8, an increase of approximately 90% over the highest release measured for the three tests completed at anticipated operating temperatures (0.51). For the remaining radionuclides whose release fractions were measured, we used the highest measured value for the tests conducted at the anticipated operating temperature and increased these values by 63% and 90%, respectively, for the tests conducted at higher than anticipated temperatures (FPFRT-4 and FPFRT-8). Table 18 lists the release fractions we selected for the nine tests, based on measurements made during the tests. The same release fractions were assumed for all isotopes of the elements shown in Table 18. For all other radionuclides not measured during the tests, release fractions of 0.1, 0.5, and 1.0 were used for solids, halogens, and noble gases, respectively. We then increased these values by 63% and 90%, respectively, for FPFRT-4 and FPFRT-8 (the release fraction for noble gases remains the same for all tests).

FPFRT-1

We assumed a wind speed of 8 mph (3.6 m s^{-1}), atmospheric stability class D, and a straight-line trajectory to the nearest offsite location in the general direction of the prevailing winds, considering the location with the highest dispersion factor reported by DOE (1991a). These assumptions resulted in the plume passing over Highway 20 (25 km downwind) and the Cellar (40 km downwind). DOE (1991a) reported a transit time to the INEEL site boundary of 1.4 hours (84 minutes), which corresponded to a wind speed of 7.9 m s^{-1}. Therefore, we also evaluated exposure to the plume at the Cellar, assuming a wind speed of 7.9 m s^{-1}. Based on these wind speeds and travel distances, the relative screening value for the FPFRT-1 release for the onsite exposure was 3.1×10^{-10} and for the offsite exposure was 5.4×10^{-8} for the 3.6 m s^{-1} wind speed and 2.5×10^{-8} for the 7.9 m s^{-1} wind speed.

FPFRT-2

We assumed a wind speed of 7.6 mph (3.4 m s^{-1}), atmospheric stability class F, and a straight-line trajectory to the nearest offsite location in the general direction of the prevailing winds, considering the location with the highest dispersion factor reported by DOE (1991a). These assumptions resulted in the plume passing over Highway 20 (onsite, 13 km downwind) and Blackfoot (offsite, 60 km downwind). DOE (1991a) reported a transit time to the INEEL site boundary of 2.2 hours (132 minutes), which corresponded to a wind speed of 7.6 m s^{-1}. Therefore, we also evaluated exposure to the plume in Blackfoot, assuming a wind speed of 7.6 m s^{-1}. Based on these wind speeds and travel distances, the relative screening value for the FPFRT-2 release for the onsite exposure was 4.7×10^{-9} and for the offsite exposure was 2.3×10^{-7} for the 3.4 m s^{-1} wind speed and 1.0×10^{-7} for the 7.6 m s^{-1} wind speed.

Table 18. Release Fractions Measured during the Fission Product Field Release Tests and Selected for Screening Associated Radionuclide Releases

Test	^{90}Sr	$^{95}Zr/Nb$	^{103}Ru	^{131}I	^{137}Cs	^{141}Ce
FPFRT-1	0.017[a]	0.0025[b]	0.037[c]	0.44[d]	0.51[e]	0.0005[f]
FPFRT-2	0.017	0.0025	0.037	0.44	0.51	0.0005
FPFRT-3	0.017	0.0025	0.037	0.44	0.51	0.0005
FPFRT-4	0.028	0.0041	0.060	0.72	0.83[g]	0.0008
FPFRT-5	0.017	0.0025	0.037	0.44	0.51	0.0005
FPFRT-6	0.017	0.0025	0.037	0.44	0.51	0.0005
FPFRT-7	0.017	0.0025	0.037	0.44	0.51	0.0005
FPFRT-8	0.032	0.0048	0.070	0.84	0.97[h]	0.003[i]
FPFRT-9	0.017	0.0025	0.037	0.44	0.51	0.0005

[a] Measured during FPFRT-1 using network air samples.
[b] Measured during FPFRT-7 using network air samples.
[c] Measured during FPFRT-5 using fractional furnace effluent sampling.
[d] Measured during FPFRT-5 using fractional furnace effluent sampling.
[e] Measured during FPFRT-3 using pre- and post-melt gamma spectrum analysis.
[f] Measured during FPFRT-7 using network air samples.
[g] Measured during FPFRT-4 using pre- and post-melt gamma spectrum analysis.
[h] Upper bound estimate for FPFRT-8 based on pre- and post-melt radiochemical analysis.
[i] Measured during FPFRT-8 using network air samples.

FPFRT-3

We assumed a wind speed of 4.5 mph (2.0 m s^{-1}), atmospheric stability class F, and a straight-line trajectory to the nearest offsite location in the general direction of the prevailing winds, considering the location with the highest dispersion factor reported by DOE (1991a). These assumptions resulted in the plume passing over Highway 20 (onsite, 25 km downwind) and the Cellar (offsite, 40 km downwind). DOE (1991a) reported a transit time to the INEEL site boundary of 10 hours (600 minutes), which corresponded to a wind speed of 1.1 m s^{-1}. Therefore, we also evaluated exposure to the plume at the Cellar, assuming a wind speed of 1.1 m s^{-1}. Based on these wind speeds and travel distances, the relative screening value for the FPFRT-3 release

for the onsite exposure was 3.9×10^{-9} and for the offsite exposure was 6.4×10^{-7} for the 2.0 m s^{-1} wind speed and 1.2×10^{-6} for the 1.1 m s^{-1} wind speed.

FPFRT-4

We assumed a wind speed of 9.8 mph (4.4 m s^{-1}), atmospheric stability class F, and a straight-line trajectory to the nearest offsite location in the general direction of the prevailing winds, considering the location with the highest dispersion factor reported by DOE (1991a). These assumptions resulted in the plume passing through Mud Lake (offsite, 45 km downwind), but it did not result in the plume passing over an onsite location with the potential for exposure. DOE (1991a) reported a transit time to the INEEL site boundary of 3.3 hours (198 minutes), which corresponded to a wind speed of 3.8 m s^{-1}. Therefore, we also evaluated exposure to the plume in Mud Lake, assuming a wind speed of 3.8 m s^{-1}. Based on these wind speeds and travel distances, the relative screening value for the FPFRT-4 release for the offsite exposure was 3.5×10^{-7} for the 4.4 m s^{-1} wind speed and 4.0×10^{-7} for the 3.8 m s^{-1} wind speed.

FPFRT-5

We assumed a wind speed of 16.1 mph (7.2 m s^{-1}), atmospheric stability class D, and a straight-line trajectory to the nearest offsite location in the general direction of the prevailing winds, considering the location with the highest dispersion factor reported by DOE (1991a). These assumptions resulted in the plume passing through Mud Lake (offsite, 45 km downwind), but it did not result in the plume passing over an onsite location with the potential for exposure. DOE (1991a) reported a transit time to the INEEL site boundary of 5.8 hours (348 minutes), which corresponded to a wind speed of 2.2 m s^{-1}. Therefore, we also evaluated exposure to the plume in Mud Lake, assuming a wind speed of 2.2 m s^{-1}. Based on these wind speeds and travel distances, the relative screening value for the FPFRT-5 release for the offsite exposure was 1.4×10^{-8} for the 7.2 m s^{-1} wind speed and 4.7×10^{-8} for the 2.2 m s^{-1} wind speed.

FPFRT-6

We assumed a wind speed of 13.7 mph (6.1 m s^{-1}), atmospheric stability class D, and a straight-line trajectory to the nearest offsite location in the general direction of the prevailing winds, considering the location with the highest dispersion factor reported by DOE (1991a). These assumptions resulted in the plume passing through Roberts (offsite, 58 km downwind), but it did not result in the plume passing over an onsite location with the potential for exposure. DOE (1991a) reported a transit time to the INEEL site boundary of 6.2 hours (372 minutes), which corresponded to a wind speed of 2.6 m s^{-1}. Therefore, we also evaluated exposure to the plume in Roberts, assuming a wind speed of 2.6 m s^{-1}. Based on these wind speeds and travel distances, the relative screening value for the FPFRT-6 release for the offsite exposure was 7.3×10^{-9} for the 6.1 m s^{-1} wind speed and 1.7×10^{-8} for the 2.6 m s^{-1} wind speed.

FPFRT-7

We assumed a wind speed of 11.0 mph (4.9 m s^{-1}), atmospheric stability class F, and a straight-line trajectory to the nearest offsite location in the general direction of the prevailing winds, considering the location with the highest dispersion factor reported by DOE (1991a). These assumptions resulted in the plume passing through Mud Lake (offsite, 45 km downwind), but it did not result in the plume passing over an onsite location with the potential for exposure. DOE (1991a) reported a transit time to the INEEL site boundary of 21.9 hours (1314 minutes), which corresponded to a wind speed of 0.6 m s^{-1}. Therefore, we also evaluated exposure to the plume in Mud Lake, assuming a wind speed of 0.6 m s^{-1}. Based on these wind speeds and travel distances, the relative screening value for the FPFRT-7 release for the offsite exposure was 5.8×10^{-8} for the 4.9 m s^{-1} wind speed and 4.8×10^{-7} for the 0.6 m s^{-1} wind speed.

FPFRT-8

We assumed a wind speed of 9.0 mph (4.0 m s^{-1}), atmospheric stability class F, and a straight-line trajectory to the nearest offsite location in the general direction of the prevailing winds, considering the location with the highest dispersion factor reported by DOE (1991a). These assumptions resulted in the plume passing through Mud Lake (offsite, 45 km downwind), but it did not result in the plume passing over an onsite location with the potential for exposure. DOE (1991a) reported a transit time to the INEEL site boundary of 14.1 hours (846 minutes), which corresponded to a wind speed of 0.9 m s^{-1}. Therefore, we also evaluated exposure to the plume in Mud Lake, assuming a wind speed of 0.9 m s^{-1}. Based on these wind speeds and travel distances, the relative screening value for the FPFRT-8 release for the offsite exposure was 1.3×10^{-7} for the 4.0 m s^{-1} wind speed and 5.9×10^{-7} for the 0.9 m s^{-1} wind speed.

FPFRT-9

We assumed a wind speed of 2.5 mph (1.1 m s^{-1}), atmospheric stability class D, and a straight-line trajectory to the nearest offsite location in the general direction of the prevailing winds, considering the location with the highest dispersion factor reported by DOE (1991a). These assumptions resulted in the plume passing over Highway 20 (onsite, 10 km downwind) and Atomic City (offsite, 19 km downwind). DOE (1991a) reported a transit time to the INEEL site boundary of 15.4 hours (924 minutes), which corresponded to a wind speed of 0.3 m s^{-1}. Therefore, we also evaluated exposure to the plume in Atomic City, assuming a wind speed of 0.3 m s^{-1}. Based on these wind speeds and travel distances, the relative screening value for the FPFRT-9 release for the onsite exposure was 3.2×10^{-9} and for the offsite exposure was 4.6×10^{-7} for the 1.1 m s^{-1} wind speed and 1.7×10^{-6} for the 0.3 m s^{-1} wind speed.

FECF Filter Break

On October 30, 1958, during decontamination operations at the Fuel Element Cutting Facility (FECF) located at the ICPP, acid fumes caused failure of the exhaust filters resulting in the release of approximately 100 Ci of aged fission products to the south of ICPP (USAEC 1960). Rich (1959) reports a total release of 1200 Ci, with 110 Ci deposited outside the perimeter fence

and the remainder deposited on the roof of the FECF and inside the perimeter fence. Additional details and discussion related to this and other particulate release problems encountered at the ICPP during this time period are provided by Hayden and Rich (1958-1959). Fuel elements being cut at the FECF at the time of the incident were reported to be approximately 1-year old (DOE 1991b). We calculated a 1-year decayed fission product inventory using RSAC-5, assuming the same reactor operating parameters used by DOE (1991b) for an MTR fuel element, and the release fractions for all fission products were proportionally adjusted to result in a total release of 100 Ci to correspond with the reported release amounts (final release fraction of 0.0186 was assumed for all fission products). This was more than a factor of 10 greater than the total release assumed by DOE (1991a), which was correlated with the exposure rate measured 3 feet above the ground at 0.6 miles (approximately 1000 meters) from the release point. We assumed the same uranium release values estimated by DOE (1991a).

We assumed a wind speed of 4.5 mph (2.0 m s^{-1}), atmospheric stability class F, and a straight-line trajectory to the nearest offsite location in the general direction of the prevailing winds, based on the transit time and location with the highest dispersion factor reported by DOE (1991a). These assumptions resulted in the plume passing over Highway 20 (onsite, 7 km downwind) and the Frenchman's Cabin location (offsite, 22 km downwind). DOE (1991a) reported a transit time to the INEEL site boundary of 1.4 hours (84 minutes), which corresponded to a wind speed of 4.4 m s^{-1}. Therefore, we also evaluated exposure to the plume in Atomic City, assuming a wind speed of 4.4 m s^{-1}. Based on these wind speeds and travel distances, the relative screening value for the FECF filter break for the onsite exposure was 6.4×10^{-7} and for the offsite exposure was 4.0×10^{-6} for the 2.0 m s^{-1} wind speed and 1.8×10^{-6} for the 4.4 m s^{-1} wind speed.

October 1959 ICPP Criticality

On October 16, 1959, at approximately 2:50 a.m., a nuclear incident occurred in a process equipment waste collection tank at ICPP. At the time of the criticality incident, the meteorological conditions that prevailed appear to have been a strong lapse condition with winds out of the north-northwest. Wind speeds at the 250-foot level varied from 14 to 31 mph (6.3 to 14 m s^{-1}) and at the 20-foot level varied from 7 to 17 mph (3 to 7.6 m s^{-1}) (Ginkel et al. 1960).

The criticality incident resulted in an estimated total of 4×10^{19} fissions (Burgus 1959, Exhibit A to Ginkel et al. 1960). We calculated a fission product inventory using the RSAC-5 computer code by correlating this to an equivalent reactor energy release of 1282 MW-s, using a conversion factor of 3.12×10^{10} fissions per W–s. Lewis (1960, Exhibit D to Ginkel et al. 1960) indicates that the reaction occurred over a period of several minutes, so we assumed a reactor power level of 4.3 MW for a period of 5 minutes (300 seconds). Limited information was available regarding the fraction of the inventory that may have been released, so we assumed release fractions of 0.1, 0.5, and 1.0 for solids, halogens, and noble gases, respectively. The inventory of ^{131}I was adjusted to approximate the calculated release, based on analyses of the stack-gas-monitor scrubber solution reported by Hayden (1959). An estimated release for ^{132}I was also reported by Hayden (1959); however, DOE (1991a) determined this value to be in error because it did not account for the fact that most of the ^{132}I in the analyzed sample would have come from the decay of ^{132}Te. Because of calculation errors related to the reported release of ^{132}I, all isotopes of iodine (except ^{131}I) were assumed to be released in quantities equal to 50% (assumed release fraction) of the RSAC-5 calculated inventory.

We assumed a wind speed of 14 mph (6.3 m s^{-1}), atmospheric stability class F, and a straight-line trajectory to the nearest offsite location in the general direction of the prevailing winds, considering the location with the highest dispersion factor reported by DOE (1991a). These assumptions resulted in the plume passing over Highway 20 (onsite, 5 km downwind) approximately 13 minutes following the release and arriving at the Frenchman's Cabin (offsite, 20 km downwind) after approximately 53 minutes. DOE (1991a) reported a transit time to the INEEL site boundary of 0.5 hours (30 minutes), which corresponded to a wind speed of 11.1 m s^{-1}. Therefore, we also evaluated exposure to the plume at the Frenchman's Cabin 30 minutes following the release, assuming a wind speed of 11.1 m s^{-1}. Based on these decay times and travel distances, the relative screening value for the 1959 ICPP criticality for the onsite exposure was 2.4×10^{-6} and for the offsite exposure was 3.7×10^{-7} for the 30 minute transit time and 5.7×10^{-7} for the 52 minute transit time.

Stationary Low-Power Reactor No. 1 Accident

The SL-1 was the smallest known power reactor when it began critical operations in August 1958. This direct cycle, natural circulation, boiling water reactor was part of the Army program to develop simple and compact package power plants to be transported by air to remote Arctic sites. At 9:02 p.m. on January 3, 1961, a nuclear excursion and steam explosion occurred in the SL-1 reactor (Gammill 1961).

The prevailing meteorological conditions at the time of the accident were characteristic of the typical conditions at this time of year. Very light surface winds were generally steady from the north to northeast with an extremely strong inversion under clear skies. In the 100-hour period following the accident, there were 98 hours with north-northeast winds at a mean speed of 7.5 mph as observed at the 250-foot level of the meteorological tower at the CFA (Islitzer 1962).

The fission product inventory in the SL-1 reactor core consisted of the radionuclides produced during the excursion and also radionuclides present as a result of previous reactor operations. The operating history of the reactor consisted of 11,000 hours for a total energy release of 932 MW-d (Gammill 1961). The reactor was then shut down on December 23, 1960 for a period of 11 days before the excursion of January 3, which resulted in a total energy release of 133 MW-s (Kunze 1962; Gammill 1961). We used the RSAC-5 computer code to calculate a fission product inventory based on operation of the reactor at a power level of 2.03 MW for 458 days, followed by a shutdown period of 11 days and the excursion power level of 88,700 MW for a period of 0.015 seconds.

Limited informatin was available regarding releases of radionuclides from this incident. Islitzer (1962) reported a total ^{131}I release of 84 Ci, based on air and vegetation samples, for the period including January 4 through February 12, 1961. This total release consisted of releases of 10 Ci on January 4, 20 Ci on January 5, 5 Ci d^{-1} between January 6 and 11, 2 Ci d^{-1} between January 12 and 17, 1 Ci d^{-1} between January 18 and 23, 0.5 Ci d^{-1} between January 23 and 29, and 0.2 Ci d^{-1} between January 30 and February 12. Gammill (1961) reports a similar total ^{131}I release of less than 80 Ci as well as release values of 0.1 Ci and 0.5 Ci for ^{90}Sr and ^{137}Cs, respectively, based on soil samples collected from the area within the SL-1 perimeter fence. We divided the total fission product inventory calculated for these radionuclides at the time of the excursion by estimated releases of 84, 0.5, and 0.1 Ci to derive release fractions for isotopes of these radionuclides. This resulted in release fractions of 0.0044 for ^{131}I, 0.00017 for ^{137}Cs, and

0.000036 for ^{90}Sr. We assumed a release fraction of 1.0 for noble gases, the same release fraction derived for ^{131}I (0.0044) for all halogens, and the same release fraction derived for ^{137}Cs (0.00017) for all solids. The reactor core consisted of 91% enriched fuel, containing 14.1 kg of ^{235}U, 20% of which was destroyed during the excursion (GE 1962). Potential uranium releases were estimated assuming uranium in the core was released in the same fraction used for solids. For our calculations, we conservatively assumed the entire 84 Ci release of ^{131}I occurred at the time of the accident.

We assumed a wind speed of 7.5 mph (3.4 m s^{-1}), atmospheric stability class F, and a straight-line trajectory to the nearest offsite location in the general direction of the prevailing winds, considering the location with the highest dispersion factor reported by DOE (1991a). These assumptions resulted in the plume passing over Highway 20 (onsite, 1.1 km downwind) approximately 5.4 minutes following the release and arriving at Atomic City (offsite, 8.5 km downwind) after approximately 42 minutes. DOE (1991a) reported a transit time to the INEEL site boundary of 1 hour (60 minutes), which corresponded to a wind speed of 2.4 m s^{-1}. Therefore, we also evaluated exposure to the plume at the Frenchman's Cabin 60 minutes following the release, assuming a wind speed of 2.4 m s^{-1}. Based on these decay times and travel distances, the relative screening value for the SL-1 accident for the onsite exposure was 1.3×10^{-5} and for the offsite exposure was 2.2×10^{-5} for the 42 minute transit time and 3.1×10^{-5} for the 60 minute transit time.

January 1961 ICPP Criticality

A nuclear incident involving an enriched uranium solution occurred in a first cycle product evaporator at the ICPP at approximately 9:50 a.m. on January 25, 1961. The period following the nuclear incident was characterized by light northerly winds and a temperature lapse up to 500 feet above the surface at 11:00 a.m. (Paulus et al. 1961). Ten-minute average wind direction and speeds were recorded at the CFA 250-foot and 20-foot wind towers and at the GRID III 140-foot wind towers and are provided by Paulus et al. (1961) for the 4-hour period following the incident. Wind speeds at the 250-foot level varied from 0 to 11 miles per hour (0 to 4.9 m s^{-1}) and at the 20-foot level varied from 0 to 9 mph (0 to 4 m s^{-1}).

The criticality incident resulted in an estimated total of 6×10^{17} fissions (Paulus et al. 1961). A fission product inventory was calculated using the RSAC-5 computer code by correlating this to an equivalent reactor energy release of approximately 20 MW-s, using a conversion factor of 3.12×10^{10} fissions per W-s. Limited information was available regarding the fraction of the inventory that may have been released, so we assumed release fractions of 0.1, 0.5, and 1.0 for solids, halogens, and noble gases, respectively.

We assumed a wind speed of 4 mph (1.8 m s^{-1}), atmospheric stability class F, and a straight-line trajectory to the nearest offsite location in the general direction of the prevailing winds, considering the location with the highest dispersion factor reported by DOE (1991a). These assumptions resulted in the plume passing over Highway 20 (onsite, 6 km downwind) approximately 56 minutes following the release and arriving at Cerro Grande (offsite, 13 km downwind) after approximately 120 minutes. DOE (1991a) reported a transit time to the INEEL site boundary of 2.3 hours (138 minutes), which corresponded to a wind speed of 1.6 m s^{-1}. Therefore, we also evaluated exposure to the plume at the Frenchman's Cabin 138 minutes following the release, assuming a wind speed of 1.6 m s^{-1}. Based on these decay times and travel

distances, the relative screening value for the 1961 ICPP criticality for the onsite exposure was 3.2×10^{-8} and for the offsite exposure was 3.6×10^{-8} for the 120 minute transit time and 3.9×10^{-8} for the 138 minute transit time.

Special Power Excursion Reactor-I Tests

The SPERT-I Pogram consisted of three series of tests concluded with a transient excursion during which fuel melting or core damage was anticipated. The tests were designed to obtain an understanding of reactor kinetic behavior and to investigate the consequences of reactor accidents. Estimation of resulting radiation exposures and fission product release to the atmosphere was also considered an integral part of the test series objectives (Bunch 1965). It was hoped that the first test would shed some light on the factors that might have been responsible for the type of destructive pressure pulses that apparently occurred during the BORAX-I excursion in 1954 and the SL-1 accident in 1961 (Miller et al. 1964). The primary objective of the second test was to study the nature of the destructive effects that could be produced following a severe power excursion in a low-enrichment oxide core. Because the second destructive test did not produce widespread cladding failure, the third test was designed to determine if substantially more severe damage would be obtained with higher fuel temperatures. The tests were conducted at the SPERT facility, near the Auxiliary Reactor Area (ARA) at the INEEL.

Weather requirements for the test consisted of lapse conditions with no rainfall, wind from the southwest (190 to 250°) between 10 and 20 mph and a 3-hour predicted persistence of these conditions after the test (Miller et al. 1964). Bunch (1965) reports slightly different weather requirements; a wind direction ranging from 200 to 240° and wind speeds ranging from 4.5 to 14 m s^{-1} (10 to 31 mph). The SPERT-I, Test No. 1 destructive test was initiated at 12:25 p.m. on November 5, 1962 after waiting approximately 2 weeks for favorable meteorological conditions. Test No. 2 was initiated at 8:15 a.m. on November 10, 1963, and Test No. 3 began at 1:14 p.m. on April 14, 1964. Actual conditions existing at the time of the three tests are provided in Table 19.

Table 19. Meteorological Conditions Existing during the SPERT-I Tests[a]

Meteorological parameter	Test No. 1	Test No. 2	Test No. 3
Mean wind speed, m s^{-1} (mph)	11.6 (25.9)	6.5 (14.5)	8.5 (19.0)
Mean wind direction	230°	230°	245°
Stability parameter (n)	0.25	0.20	0.20

[a]Source: Bunch (1965)

SPERT-I, No. 1 Test

The SPERT-I No. 1 destructive test involved reactor operation for 3.2 msec with a nuclear energy release of 30.7 MW-s (Bunch 1965). In addition to the destructive test, the operational history of the reactor core included 50 non-destructive runs (Miller et al. 1964). Each of these runs was modeled separately according to the total energy and operating times provided by Miller et al. (1964). Total energy was not specified for several runs, so we assumed the average energy of the runs for which energy levels were specified. For the destructive test, we assumed a power

level of 9,600 MW for 3.2 msec and used the RSAC-5 computer code to calculate the fission product inventory for the reactor core based on these operating parameters.

A violent explosion occurred immediately after the final power excursion, during which complete fuel plate melting was experienced in approximately 8% of the core, with partial melting in approximately 35% of the core. It was reported that "it appears that those isotopes which were collected were released as gases. No solid products were collected." A fractional release for noble gases was estimated to be 0.07, and no halogens were identified by gamma spectrometry (Miller et al. 1964). However, because they are normally expected to contribute significantly to the possible hazards associated with fission product releases, calculations were made to estimate the maximum possible release for two isotopes of iodine. Bunch (1965) estimated that the maximum fractional releases for ^{131}I and ^{135}I were 0.00006 and 0.000003, respectively. Miller et al. (1964) reports that less than 0.01% of the iodines was released to the atmosphere. Based on this information, we assumed release fractions of 0.1 for noble gases, 0.0001 for halogens, and 0.00001 for solids. The same release fraction used for the solids was also assumed for the uranium in the fuel of the reactor core.

We assumed a wind speed of 25.9 mph (11.6 m s^{-1}), atmospheric stability class D, and a straight-line trajectory to the nearest offsite location in the general direction of the prevailing winds, considering the location with the highest dispersion factor reported by DOE (1991a). These assumptions resulted in the plume passing over Highway 33 (onsite, 35 km downwind) approximately 50 minutes following the release and arriving at Monteview (offsite, 44 km downwind) after approximately 63 minutes. DOE (1991a) reported a transit time to the INEEL site boundary of 1.1 hours (66 minutes), which corresponded to a wind speed of 11.1 m s^{-1}. Therefore, we also evaluated exposure to the plume in Monteview 66 minutes following the release, assuming a wind speed of 11.1 m s^{-1}. Based on these decay times and travel distances, the relative screening value for the SPERT-I No. 1 test for the onsite exposure was 9.1×10^{-12} and for the offsite exposure was 1.57×10^{-11} for the 63 minute transit time and 1.59×10^{-11} for the 66 minute transit time.

SPERT-I, No. 2 Test

The SPERT-I No. 2 test involved reactor operation for 2.2 msec with a nuclear energy release of 155 MW-s (Bunch 1965). There is no indication of a previous operating history for this test. We assumed a power level of 70,000 MW for 2.2 msec and used the RSAC-5 computer code to calculate the fission product inventory for the reactor core based on these operating parameters.

Only slight damage to the reactor core occurred during the second test, with two fuel pins being ruptured. A fractional release of 0.0002 was estimated for noble gases, and a maximum release fraction of 0.0001 was estimated for halogens (Bunch 1965). Because of the small amount of fuel damage and the scrubbing action of the water in the reactor core, the release of fission product solids and uranium in the fuel was likely negligible. Based on this information, we assumed release fractions of 0.0002 for noble gases, 0.0001 for halogens, and 0 for solids.

We assumed a wind speed of 14.5 mph (6.5 m s^{-1}), atmospheric stability class F, and a straight-line trajectory to the nearest offsite location in the general direction of the prevailing winds, considering the location with the highest dispersion factor reported by DOE (1991a). These assumptions resulted in the plume passing over Highway 20 (onsite, 20 km downwind) approximately 51 minutes following the release and arriving at the Cellar (offsite, 35 km

downwind) after approximately 90 minutes. DOE (1991a) reported a transit time to the INEEL site boundary of 3.9 hours (234 minutes), which corresponded to a wind speed of 2.5 m s^{-1}. Therefore, we also evaluated exposure to the plume at the Cellar 234 minutes following the release, assuming a wind speed of 2.5 m s^{-1}. Based on these decay times and travel distances, the relative screening value for the SPERT-I No. 2 test for the onsite exposure was 2.5×10^{-12} and for the offsite exposure was 3.1×10^{-12} for the 90 minute transit time and 6.1×10^{-12} for the 234 minute transit time.

SPERT-I, No. 3 Test

The SPERT-I No. 3 test involved reactor operation for 1.55 msec with a nuclear energy release of 165 MW-s (Bunch 1965). There is no indication of a previous operating history for this test. We assumed a power level of 106,000 MW for 1.55 msec and used the RSAC-5 computer code to calculate the fission product inventory for the reactor core based on these operating parameters.

The third test resulted in limited damage to the reactor core, with two fuel pins rupturing before the time of peak power. A fractional release of 0.0006 was estimated for noble gases, and a maximum release fraction of 0.0001 was estimated for halogens (Bunch 1965). Because of the small amount of fuel damage and the scrubbing action of the water in the reactor core, the release of fission product solids and uranium in the fuel was likely negligible. Based on this information, we assumed release fractions of 0.0006 for noble gases, 0.0001 for halogens, and 0 for solids.

We assumed a wind speed of 19.0 mph (8.5 m s^{-1}), atmospheric stability class D, and a straight-line trajectory to the nearest offsite location in the general direction of the prevailing winds, considering the location with the highest dispersion factor reported by DOE (1991a). These assumptions resulted in the plume arriving at Mud Lake (offsite, 43 km downwind) after approximately 84 minutes, but it did not resulted in the plume passing over an onsite location with the potential for exposure. DOE (1991a) reported a transit time to the INEEL site boundary of 1.2 hours (72 minutes), which corresponded to a wind speed of 10.0 m s^{-1}. Therefore, we also evaluated exposure to the plume at Mud Lake 72 minutes following the release, assuming a wind speed of 10.0 m s^{-1}. Based on these decay times and travel distances, the relative screening value for the SPERT-I No. 3 test for the offsite exposure was 6.6×10^{-13} for the 72 minute transit time and 7.2×10^{-13} for the 84 minute transit time.

SNAP10A Transient Program

The AEC initiated a nuclear safety program to evaluate the hazards associated with using nuclear reactors for aerospace auxiliary power systems. The portion of the program concerned with determining the kinetic behavior of the SNAP 10A/2 reactors and the consequences of certain nuclear accidents involving these reactors was designated as SNAPTRAN. The SNAP 10A/2 reactors were approximately 9 inches in diameter by 12 inches long and were composed of a NaK-cooled core containing 37 rods of fully-enriched uranium in a zirconium-hydride matrix (Berta 1967, 51363).

The SNAPTRAN tests were designed to investigate the consequences of a nuclear accident resulting from two potentially hazardous situations: (1) the immersion of the reactor core in water or moist earth, and (2) the accidental rotation of the control drums into the reactor during

assembly or launch. The SNAPTRAN-3 Program investigated the consequences resulting from the water immersion accident, and the SNAPTRAN-2 and SNAPTRAN-1 Programs investigated the consequences of accidental rotation of the control drums into the reactor. Partial damage kinetic testing was conducted in the SNAPTRAN-1 test program, and a total destructive test was conducted in the SNAPTRAN-2 test program (Berta 1967). Because the SNAPTRAN-1 test was designed to study reactor kinetics in the nondestructive region, atmospheric releases of fission products were not significant. The tests were conducted at TAN. An extensive radiological and meteorological network was established to carefully monitor the radiological consequences of the tests.

SNAPTRAN-3 Test

The SNAPTRAN-3 test was initiated at 11:44 a.m. MST on April 1, 1964. Weather requirements for the test consisted of lapse conditions with no rainfall, to persist a minimum of 3 hours after the test, and wind from the southwest (180 to 240°) between 10 and 30 mph (Cordes et al. 1965). Actual conditions at the time of the SNAPTRAN-3 test are provided in Table 20.

Table 20. Meteorological Conditions Existing during the SNAPTRAN-3 Test[a]

Instrument location	Wind direction	Wind speed (mph)
IET 20-foot tower	203°	22.2
IET 200-foot tower	208°	25.5
Station 179[a]	209°	17.7
Station Monteview	197°	18.6
Tetroon[b] (released after test)	209°	27.9

[a] Along Idaho Highway 28

[b] Balloon-like device used to collect meteorological data

The SNAPTRAN-3 test involved reactor operation at a power level of 30,000 MW for 1.5 msec (Cordes et al. 1965). We used the RSAC-5 computer code to calculate the fission product inventory for the reactor core based on these reactor operating parameters. It was estimated that greater than 99% of the fission product inventory was retained in the surrounding water and reactor fuel remains. No airborne iodine was detected, so it was presumed that any halogens escaping the fuel were retained in the water. We also assumed that the water retained any particulate radionuclides, including uranium from the fuel elements, and prevented them from being released. Cordes et al. (1965) reports that "the upper limit of noble gas release...is four percent." We assumed a release fraction of 0.04 for noble gases, a conservative release fraction of 0.02 for halogens (because none were detected), and a release fraction of 0.0 for particulate radionuclides.

We assumed a wind speed of 22.2 mph (9.9 m s^{-1}), atmospheric stability class D, and a straight-line trajectory to the nearest offsite location in the general direction of the prevailing winds, considering the location with the highest dispersion factor reported by DOE (1991a). These assumptions resulted in the plume passing over Highway 28 (onsite, 12 km downwind) approximately 20 minutes following the release and arriving at Reno Ranch (offsite, 17 km downwind) after approximately 29 minutes. DOE (1991a) reported a transit time to the INEEL

site boundary of 0.6 hours (36 minutes), which corresponded to a wind speed of 7.9 m s^{-1}. Therefore, we also evaluated exposure to the plume in Reno Ranch 36 minutes following the release, assuming a wind speed of 7.9 m s^{-1}. Based on these decay times and travel distances, the relative screening value for the SNAPTRAN-3 test for the onsite exposure was 8.0×10^{-11} and for the offsite exposure was 7.8×10^{-11} for the 29 minute transit time and 8.5×10^{-11} for the 36 minute transit time.

SNAPTRAN-2 Test

The SNAPTRAN-2 test began at 9:51 a.m. MST on January 11, 1966. Weather requirements for the test consisted of neutral to light lapse conditions with no imminent rainfall, to persist a minimum of 30 minutes after the test, and wind from the southwest (180 to 240°) between 3 and 18 m s^{-1} (7 and 40 mph) (Cordes et al. 1967). Actual conditions at the time of the SNAPTRAN-2 test consisted of winds out of the south to southwest, and at 9:40 a.m. the wind was reported at 5 m s^{-1} (11 mph) (Cordes et al. 1967).

The SNAPTRAN-2 test involved reactor operation at a power level of 36,000 MW for 1.5 msec (Cordes et al. 1967). We used the RSAC-5 computer code to calculate the fission product inventory for the reactor core based on these operating parameters. Cordes et al. 1967) reports fission product release fractions of 0.75 for noble gases, 0.70 for iodines, 0.45 for tellurium, and 0.04 for solids. We used these release fractions, assumed the same release fraction (0.70) for all halogens, and assumed the same release fractions for the uranium in the fuel elements that was reported for the solids. The reactor core consisted of 93% enriched fuel, containing 4.75 kg of ^{235}U. The higher release fractions for the SNAPTRAN-2 test were related to more complete fragmentation of the fuel matrix than occurred during SNAPTRAN-3. This presumably did not occur during the water immersion test (SNAPTRAN-3) because the cooling action of the water limited the fragmentation of the fuel matrix, and the water itself retained much of the material that was released from the fuel.

We assumed a wind speed of 11 mph (5 m s^{-1}), atmospheric stability class F, and a straight-line trajectory to the nearest offsite location in the general direction of the prevailing winds, considering the location with the highest dispersion factor reported by DOE (1991a). These assumptions resulted in the plume passing over Highway 28 (onsite, 10 km downwind) approximately 33 minutes following the release and arriving at Monteview (offsite, 17 km downwind) after approximately 57 minutes. DOE (1991a) reported a transit time to the INEEL site boundary of 2.9 hours (174 minutes), which corresponded to a wind speed of 1.6 m s^{-1}. Therefore, we also evaluated exposure to the plume in Monteview 174 minutes following the release, assuming a wind speed of 1.6 m s^{-1}. Based on these decay times and travel distances, the relative screening value for the SNAPTRAN-2 test for the onsite exposure was 4.0×10^{-8} and for the offsite exposure was 4.1×10^{-8} for the 57 minute transit time and 1.1×10^{-7} for the 174 minute transit time.

Experimental Release Tests

There have been a number of intentional release tests at the INEEL designed to study a number of parameters related to the movement of radionuclides in the environment. These are

briefly described in the sections below, followed by an evaluation of the potentially importance of exposure as a result of the materials released during the tests.

Controlled Environmental Radioiodine (Release) Tests

The CERT releases were conducted from May 27, 1963 through December 1977. The primary initial objectives of the releases were to establish relationships between the amounts of radioiodine in different environmental media. Specifically, relationships between air and soil and vegetation, vegetation and milk, and milk and human thyroids were studied. The tests involved releases of both elemental and methyl radioiodine ranging in amount from 0.05 to 8 Ci. Most of the releases occurred at the Experimental Dairy Farm, also know as the Experimental Field Station and located to the northeast of the ICPP. Other releases occurred at the ICPP, ARA, NRF, and CFA areas. The name was changed in 1968 to Controlled Environmental Release Test to reflect the release and study of additional radionuclides, such as cesium, cerium, potassium, and krypton. Additional information regarding this test series is provided by Hawley (1964), Bunch (1966 and 1968), and Zimbrick and Voillequé (1969).

Relative Diffusion Tests

The Relative Diffusion Test releases occurred between November 30, 1967 and October 1, 1969. Four tests were conducted at GRID III and involved the release of both methyl and elemental radioiodine. Quantities released varied from 1 to 6 Ci. Details regarding these releases are limited, but some additional information can be found in DOE (1991b).

Experimental Cloud Exposure Study

The Experimental Cloud Exposure Study (EXCES) releases were conducted from May 3, 1968 through April 24, 1970 at GRID III. Tests during 1968 and 1969 consisted of ^{133}Xe releases ranging from 32 to 600 Ci, and tests in 1970 consisted of ^{24}Na releases ranging from 6.6 to 120 Ci. The primary objectives for the tests included measuring total exposure at several downwind distances; determining dimensions of the plumes; document the release rate and height, wind speed, and temperature; and measuring the gamma energy spectrum at one or more points during the release. Details regarding these releases are limited, but some additional information can be found in DOE (1991b). Ruhter (1970) provides information regarding the safety planning and preparation carried out in support of the EXCES ^{24}Na releases. Releases were planned to occur during meteorological conditions characterized by winds out of the southwest to minimize potential onsite and offsite exposure and also to ensure the cloud passing over preset instrumentation. Voillequé (1969) discusses an outline of plans for the EXCES ^{133}Xe release tests, including a discussion of the general objectives and procedures associated with the tests.

Long Distance Diffusion Tests

The Long Distance Diffusion Test (LDDT) releases occurred between March 3, 1971 and September 22, 1972. The releases occurred at the ICPP and involved 3.7 to 4.4 Ci amounts of methyl radioiodine and 1000 Ci of ^{85}Kr. The primary objective of the tests was to evaluate

mesoscale atmospheric dispersion of non-depositing tracer gases at the INEEL. Dickson and Voillequé (1972) provides additional details regarding these tests. Voillequé (1971) briefly discusses the first two tests in the Phase I series, which consisted of releasing 4.4 and 4.2 Ci of ^{131}I-labeled CH_3I during 65-minute periods on March 3, 1971 and August 31, 1971, respectively.

Evaluation of Potential Importance of the Release Tests

We initially evaluated the relative importance of these release tests by treating the entire series of releases from the four release tests (CERT, RDT, EXCES, and LDDT) in a very simplified manner for comparison to the relative screening values determined for other site releases. Each release was treated very conservatively, assuming class F atmospheric stability and a constant wind out of the northwest at a speed of 2 m s^{-1}, taking the plume in a straight-line trajectory to the nearest offsite person, Atomic City. A total integrated concentration was calculated as with the other episodic evaluations, and the appropriate total screening factors were used to estimate a relative screening value for each release. These values were compared with the values obtained for other tests to determine which releases warranted further investigation. The highest relative screening value of 1.6×10^{-6} was for the CERT #1 release of 1 Ci ^{131}I, followed by several releases with screening values greater than 1.0×10^{-7}.

We then refined our calculations, using existing wind speeds and directions, if known, as well as a more appropriate atmospheric stability, depending on the time of year and time of day as well as existing meteorological conditions. Many of the releases were initiated only when the wind was blowing from the southwest, which would carry the plume across the uninhabited site toward the nearest offsite person in Mud Lake. Based on these more realistic conditions, we again calculated relative screening values. The highest value of 7.4×10^{-7} was for the RDT-4 release, followed by six releases with a screening value greater than 1.0×10^{-7}, all of which involved releases of ^{131}I. Four of these seven tests involved the release of methyl radioiodine, which is virtually non-depositing, so the air to vegetation pathway was not an important milk contamination pathway. Because the majority of the dose from ^{131}I comes from ingestion, the relative screening values calculated assuming the releases consisted of elemental radioiodine were likely erroneously high. Additionally, more realistic and representative information was not obtained for five of the seven releases with relative screening values greater than 1.0×10^{-7} (i.e., they were evaluated assuming winds out of the northwest). It is likely that these releases were restricted to times during which winds were blowing out of the southwest because the data collection grid for the majority of these test releases was situated to the northeast of the release point. This would further reduce the screening values. However, as mentioned previously, available data regarding these tests are limited, and records of meteorological conditions during some of the tests were not located. The release test with the highest screening value evaluated using assumed meteorological conditions was the RDT-4 release, which had a relative screening value of 7.4×10^{-7}. The release test with the highest screening value evaluated using meteorological conditions existing at the time of the test was the CERT #11 release, which had a relative screening value of 2.6×10^{-7}.

Keeping this information in mind, it is evident that the highest screening values for these release tests (many of which were likely erroneously high) were significantly lower than the values for the most important (i.e., highest screening values) release events already evaluated. For comparison, a RaLa run on May 28, 1958 released nearly 50 Ci ^{131}I. Given the relative magnitude

of these test releases of ^{131}I, we determined that it was not necessary to further evaluate the potential exposure implications associated with them.

Other Episodic Releases

We focused our analysis of releases to this point on significant accidental releases and experimental release tests that required substantial reconstruction of the events based on reactor operating histories and also on several series of planned experimental release tests. A number of additional releases that could be considered episodic have occurred at the INEEL, most of which have resulted from operational problems. DOE (1991a) discusses several of these, but we have also identified a number of other releases throughout the course of our searches and reviews of the Phase I and Task Order 6 databases as well as various accident and release summary documents. These releases are discussed and evaluated in more detail in the following sections.

We evaluated these releases in a manner similar to that used for the experimental release tests, discussed above. We calculated onsite exposure relative screening values assuming a distance of 6 km to the nearest onsite location, which is consistent with the approximate distance to the nearest publicly accessible highway from ICPP and TRA, from which the majority of the releases occurred.

Releases Discussed in the INEEL Historical Dose Evaluation

A number of releases discussed by DOE (1991a) are not specifically evaluated as episodic releases. There are several reasons for this decision, including insufficient available information to reconstruct a source term, routine monitoring in place during the time of the release, a decision to evaluate the release as part of a routine annual release, and information suggesting that a particular release was of such a nature that it would not be expected to significantly impact offsite doses. This section evaluates the potential offsite exposures associated with these releases. When sufficient information was available regarding the quantities and specific radionuclides involved, we evaluated these releases in a manner similar to that used for the experimental release tests. In cases where we were unable to obtain detailed information regarding the precise quantity of a given release, we examined the details related to the release and compared the details to other similar releases.

EBR-I Core Meltdown. The EBR was operated early in 1954 intermittently at power levels up to 1150 kW, which was the maximum authorized power level. Core meltdown occurred on November 29, 1955, and involved the melting of 40 to 50% of the core (DOE 1991a). The low power of the reactor, low concentrations detected in the building at the time of the meltdown, the number of total fissions reported to occur (4.7×10^{17}) (Marter 1965), and the generally low releases from this type of reactor suggest that this test would have resulted in releases of lesser magnitude than other evaluated episodic releases. Therefore, the EBR-I core meltdown was not evaluated further.

RaLa Iodine Releases. DOE (1991a) evaluated releases related to the RaLa campaign carried out at the ICPP as routine annual releases. However, there are a number of days during which significant amounts of iodine were emitted. Additionally, significant quantities of iodine

were released during the various RaLa runs, which occurred over periods of weeks and even months. We evaluated both the highest daily releases as well as the highest total releases associated with specific RaLa runs to enable comparison to other episodic releases occurring on a specific day and those occurring over a period of several days or more (e.g., IET releases). Although reported releases included ^{131}I, ^{132}I, and other beta-emitting radionuclides, we focused our evaluation here on releases of ^{131}I because it is the isotope that is potentially most important with regard to exposure to members of the public.

We tabulated daily release estimates reported by Hayden (1957-1963). We compared the summed daily releases based on these data to the release values reported by DOE (1991a) for each individual RaLa run. There were some discrepancies in the data, likely a result of differences in start and stop times, summation methodology, and the fact that the data reported by Hayden (1957-1963) included all releases from the ICPP (i.e., not just those related to RaLa). In an effort to be conservative, we based our evaluation on the highest RaLa run release estimates.

The largest daily release of ^{131}I occurred on May 28, 1958, and amounted to 49.5 Ci ^{131}I. We evaluated this release assuming a wind speed of 2 m s^{-1}, atmospheric stability class F, and a straight-line trajectory to Atomic City, the nearest offsite exposure location. Based on these assumptions, the relative screening value for the onsite location was 1.3×10^{-7} and for the offsite location was 7.1×10^{-6}. Significant quantities of ^{131}I were released on a number of other days; those days where reported releases exceeded 30 Ci are also included in the Excel spreadsheet.

The largest release associated with an individual RaLa run occurred during Run #2, and it amounted to a total release of 351 Ci ^{131}I between February 24, 1957 and March 30, 1957. We evaluated this release assuming a wind speed of 2 m s^{-1}, atmospheric stability class D, and a straight-line trajectory to Atomic City, the nearest offsite exposure location. Based on these assumptions, the relative screening value for the onsite location was 1.6×10^{-7} and for the offsite location was 7.8×10^{-6}. Significant quantities of ^{131}I were released during several other RaLa runs; those runs where reported releases exceeded 200 Ci are also included in the Excel spreadsheet.

1958 Iodine Release. Approximately 1 Ci of ^{131}I was released to the atmosphere during a routine waste transfer operation at the ICPP on March 20, 1958 (ERDA 1977, Hayden 1958, and Rigstad 1958). We evaluated this release assuming a wind speed of 2 m s^{-1}, atmospheric stability class F, and a straight-line trajectory to Atomic City, the nearest offsite exposure location. Based on these assumptions, the relative screening value for the onsite location was 2.6×10^{-9} and for the offsite location was 1.4×10^{-7}.

BORAX-IV Test. Between March 11 and 27, 1958, the BORAX-IV reactor was intentionally operated at a power level of 2.4 MW with a large number of defective fuel elements. During the tests, fission products leaked from the fuel causing high radiation and building contamination, but environmental releases were primarily limited to cleanup operations (ERDA 1977). Monitoring activities indicated ^{138}Cs to be the predominant isotope released, and it was reported that no exposure to personnel occurred beyond the project area (USAEC 1959). We have been unable to locate sufficient information to reconstruct and estimate releases from this test, but the reactor power levels and description of events indicate that this test would have resulted in releases of lesser magnitude than other evaluated episodic releases. Therefore, the BORAX-IV test was not evaluated further.

Solvent Burner Release. Approximately 0.25 Ci of long-lived particulate activity was released from the ICPP solvent burner via the main stack in September 1958 (ERDA 1977). We evaluated this release assuming a wind speed of 2 m s^{-1}, atmospheric stability class F, and a straight-line trajectory to Atomic City, the nearest offsite exposure location. We also assumed the release was comprised entirely of ^{90}Sr. Based on these assumptions, the relative screening value for the onsite location was 2.4×10^{-8} and for the offsite location was 2.4×10^{-7}.

Collection Tank Release. Approximately 1 Ci of radioactive noble gases and iodine was released to the atmosphere from a liquid waste tank at the ICPP in December 1958 as a result of a leaking flange (ERDA 1977). We evaluated this release assuming a wind speed of 2 m s^{-1}, atmospheric stability class F, and a straight-line trajectory to Atomic City, the nearest offsite exposure location. We also assumed the release was comprised entirely of ^{131}I. Based on these assumptions, the relative screening value for the onsite location was 2.6×10^{-9} and for the offsite location was 1.4×10^{-7}.

ICPP Plutonium Release. Approximately 105 mCi of plutonium were released to the atmosphere at the ICPP between July 9 and 11, 1959 (USAEC 1960). The release resulted from the burning of plutonium-contaminated waste solvent and was believed to be emitted from ventilation ports in the furnace box of the burner and the exhaust line venturi, which is approximately 20 feet above the ground level. Ground surveys for alpha contamination were conducted in the vicinity of the burner, and the only positive results were found 75 feet from the burner building. We evaluated this release assuming a wind speed of 2 m s-1, atmospheric stability class F, and a straight-line trajectory to Atomic City, the nearest offsite exposure location. Based on these assumptions, the relative screening value for the onsite location was 1.3 $\times 10^{-6}$ and for the offsite location was 5.4×10^{-7}.

Organic Moderated Reactor Experiment Solvent Burning Experiment. On November 16, 1960, an experiment was conducted to determine the feasibility of open-air burning of contaminated solvents, accumulated at the Organic Moderated Reactor Experiment (OMRE) facility. Approximately 400 gallons of liquid composed of diesel oil, xylene, methyl-chloroform, and a small amount of water were placed in an open vessel and ignited. Lapse conditions accompanied by a 25-mph wind existed at the time of the test. The radioactivity concentrations in the xylene were highest and reported to be 0.017 µCi ml^{-1}. The specific contaminants and their percentages were reported to be ^{54}Mn (60%), ^{60}Co (30%), and ^{59}Fe (10%) (ERDA 1977).

We evaluated this release assuming a wind speed of 2 m s^{-1}, atmospheric stability class D, and a straight-line trajectory to Cerro Grande, the nearest offsite exposure location. We assumed that the entire 400 gallons had the radioactivity concentration reported for xylene and that ^{60}Co, which has the highest screening factor of the three reported contaminants, was the only radionuclide present. If everything present was volatilized, a total release of 0.026 Ci could have been released to the atmosphere. Based on these assumptions, the relative screening value for the onsite location was 2.6×10^{-10} and for the offsite location was 4.9×10^{-8}.

ETR Sight Glass Incident. On December 12, 1961, ETR experienced fission breaks in six fuel elements as a result of primary coolant flow blockage to the northeast quadrant of the core

(Warzel 1961). The obstruction was caused by a Plexiglas sight glass inadvertently left in the reactor vessel during the previous shutdown. The fission break itself resulted in negligible exposure to personnel, but there was an immediate release of radioactivity to the environment through the ETR stack. This immediate release consisted of 0.4 Ci of particulate fission gas daughter activity and 6.0 Ci of fission gas activity (Keller 1962). Additionally, Rich (1962) reports an increase in the stack activity discharge rate of 50 times normal during reactor operation immediately following the accident, with a continued increased rate for at least 10 days. Keller (1962) reports an increased discharge rate for the 3-month period following the incident of approximately 10 times the rate in existence during the 3-month period preceding the break.

We evaluated the release immediately following the incident assuming a wind speed of 2 m s^{-1}, atmospheric stability class F, and a straight-line trajectory to the Frenchman's Cabin, the nearest offsite exposure location. We assumed the particulate activity was composed entirely of ^{138}Cs and the fission gas activity consisted entirely of ^{88}Kr. Based on these assumptions, the relative screening value for the onsite location was 3.8×10^{-10} and for the offsite location was 9.2×10^{-11}. The increased discharge rates during the months following this incident were not evaluated here as an episodic release but are instead considered in the routine release evaluation.

MTR Fuel Melt Incident. On November 13, 1962 the MTR was shut down by a scram resulting from flow restriction and an ensuing fission break in a standard fuel assembly (Dykes et al. 1965; Gibson et al. 1963). Radiation levels forced a 15-minute evacuation of the Reactor and Wing Buildings. Subsequent inspection of the assembly revealed a piece of debris blocking the flow in approximately 40% of the fuel assembly channels. The debris was later identified as a piece of gasket material that had broken off the seal tank floating roof seal. No major air activity problems were encountered, although a significant and expected rise in the stack effluent did occur (Smith 1962). Airborne effluent releases resulting from this fuel melt incident were recorded by the MTR stack exhaust monitor and are included in the routine release evaluation.

ICPP Waste Tank Farm Incident. On May 10, 1964, a release of fission products occurred at the ICPP Waste Tank Farm during a steam-flushing operation. During the operation, a leak developed in a hose coupling releasing contaminated fluid and steam, which was rapidly dispersed by a 30 mph wind blowing from the southwest (USAEC 1965; DOE 1991b). Contamination, consisting primarily of aged fission products, was spread over an area of approximately 3 acres inside the plant fence and over an area of approximately 10 acres outside the plant fence. DOE (1991a) reported the estimated activity that remained airborne to be about 70 mCi, with a total release of about 2% of the quantity of radionuclides released from the FECF filter break, which consisted of similar radionuclides. Because of the relatively small release associated with this event, no contamination was detected beyond the 10-acre area outside the plant fence, and the released material was dispersed to the northeast this release was not evaluated further.

ETR Fuel Melt Incident. On February 20, 1967, the ETR reactor was shut down because of excess activity in the M-16 area of the core. Subsequent investigation showed that fuel element E-018-D had failed due to coolant channel blockage caused by a piece of adhesive tape that had inadvertently been left in the reactor. De Boisblanc (1969) states that "the high viscosity of the molten core did not allow the loss of appreciable quantities of fission products and kept the core

relatively intact while the more fluid molten cladding material was swept away by the coolant flow." Francis and Tingey (1967) reports a loss of 7.498 g of fuel and that "the fission product loss to the reactor environment was less than expected, indicating that a considerable amount of fission products was retained within the fuel element." In light of these reports and because the release was monitored by the ETR stack monitor, this event was not evaluated as a separate episodic release; however, it is included in the routine release evaluation.

EBR-II Sodium Release. In February 1968, approximately 80 gallons of sodium were inadvertently released and immediately ignited in the Sodium Boiler Plant Building control room at EBR-II (ERDA 1977). The sodium contained approximately 4 mCi of ^{24}Na, and it was estimated that 0.4 μCi was released to the atmosphere. We evaluated this release assuming a wind speed of 2 m s^{-1}, atmospheric stability class F, and a straight-line trajectory to the Building, the nearest offsite exposure location. Based on these assumptions, the relative screening value for the onsite location was 7.6×10^{-17} and for the offsite location was 2.3×10^{-16}.

TSF Evaporator Release. In September 1971, an accidental airborne release occurred from the TSF liquid waste evaporator (ERDA 1977). The release was estimated to consist of approximately 266 μCi of ^{137}Cs, 0.0142 μCi of ^{90}Sr, and 0.0142 μCi of ^{90}Y. We evaluated this release assuming a wind speed of 2 m s^{-1}, atmospheric stability class F, and a straight-line trajectory to the Building, the nearest offsite exposure location. Based on these assumptions, the relative screening value for the onsite location was 4.0×10^{-13} and for the offsite location was 2.5×10^{-10}.

1972 Particulate Release. ERDA (1977) reports that approximately 1 Ci of Ru-106 was released from the main stack at the ICPP in January 1972. Black and Chamberlain (1972) report daily releases from the ICPP stack during January 1972, and the values were not consistent with the 1 Ci release reported by ERDA (1977) for January 1972. It was not clear, though, whether this value was related to a specific daily release or for the entire month. To be conservative, we based our analysis on the data reported by Black and Chamberlain (1972). WCF was shut down on January 3 for replacement of a leaking valve, and operation was resumed on January 23. The waste evaporator was operated and intermittent decontamination and maintenance of the WCF were in progress during the down period (Buckham 1972). The releases apparently resulted from failure of a filter (Wehman 1972) and occurred throughout the down period.

Generally, the majority of activity in the releases and in collected particles is comprised of ^{106}Ru, but on January 4, the ratio of ^{106}Ru activity to ^{137}Cs activity was reported to be 0.8. The highest reported daily release occurred on January 9, 1972, when a total of 2.2 Ci was released. The ratio of ^{106}Ru to ^{137}Cs activity was not reported that day, so we assumed the lowest ratio reported (0.8), which corresponded to a release of 1.0 Ci of ^{106}Ru and 1.2 Ci of ^{137}Cs. We evaluated this release assuming a wind speed of 2 m s^{-1}, atmospheric stability class F, and a straight-line trajectory to Atomic City, the nearest offsite exposure location. Based on these assumptions, the relative screening value for the onsite location was 2.4×10^{-8} and for the offsite location was 7.7×10^{-7}.

1978 ICPP Criticality. A criticality event occurred in the first-cycle tributylphosphate extraction system in the CPP-601 process building at the ICPP on October 17, 1978. It was stated

that the incident resulted in no personnel injury, no onsite or offsite contamination, and no damage to equipment or property, and the plume traveled over uninhabited areas to the southwest of the Site (Casto 1980). The criticality caused approximately 3×10^{18} fissions of ^{235}U, which is more than an order of magnitude lower than the number of fissions resulting from the 1959 ICPP criticality. In addition, the APS at ICPP, which became operational in 1975 and significantly reduced particulate emissions, filtered all releases associated with the criticality event in 1978. Reported airborne releases from ICPP for the entire year of 1978 are also generally an order of magnitude lower for the noble gas isotopes of krypton and xenon than for our reconstructed 1959 ICPP criticality release (DOE 1979). The combination of these facts suggests that releases associated with the 1978 ICPP criticality were insignificant by comparison to other similar events, such as the 1959 and 1961 ICPP criticalities, which have been evaluated in detail.

LOFT LP-FP-2 Test. The final experiment in a series of eight tests conducted under the support and direction of the Organization for Economic Cooperation and Development was carried out on July 9, 1985. The LOFT LP-FP-2 test was designed to simulate a small break loss of coolant accident, similar to the one experienced at Three-Mile Island in March 1979. The test resulted in a release of fission products to the primary coolant system. The water and fission products were then expelled from the primary coolant system during blowdown and captured by the blowdown suppression tank. Leakage from the fission product monitoring system and the primary coolant system during the 60-day period following the test allowed fission products to enter the reactor building. It was determined that release of the material to the environment would not violate LOFT Technical Specifications or radiation protection standards in place at the time, so the reactor building ventilation system was operated to evacuate the building air through a filtered and monitored pathway to the environment (Carboneau 1987).

Over the 2-month period following the test, 8780 Ci of noble gases and 0.09 Ci of iodine were released to the environment (Hoff et al. 1986). These values are consistent with the reported release of 12–13 mCi of iodine released as of July 11, 1985 (Stachew 1985). We evaluated this release assuming a wind speed of 2 m s^{-1}, atmospheric stability class D, and a straight-line trajectory to the Building, the nearest offsite exposure location. We also assumed the noble gas release consisted entirely of ^{88}Kr, which has the largest screening factor for any of the noble gas isotopes, and the radioiodine release consisted entirely of ^{131}I. Based on these assumptions, the relative screening value for the onsite location was 8.7×10^{-8} and for the offsite location was 3.2×10^{-8}.

1988 Ruthenium Release. Approximately 0.17 Ci of Ru-106 was released from the main stack at the ICPP on October 30, 1988 (Hoff et al. 1989, Volpe et al. 1988, and Mikkola 1988). We evaluated this release assuming a wind speed of 2 m s^{-1}, atmospheric stability class F, and a straight-line trajectory to Atomic City, the nearest offsite exposure location. Based on these assumptions, the relative screening value for the onsite location was 3.7×10^{-9} and for the offsite location was 5.9×10^{-9}.

Releases Identified in Reviews of INEEL Databases, Annual Reports, Environmental Monitoring Reports, and Various Accident and Release Summary Documents

In addition to the releases evaluated by DOE (1991a) we also identified a number of other releases that could be considered episodic throughout the course of our searches and reviews of the Phase I and Task Order 6 databases as well as annual reports, environmental monitoring reports, and various accident and release summary documents. Reviewed documents included Hayes (1956 and 1959), USAEC (1959, 1960, 1961, 1963, 1965, 1967a, 1967b, and 1975), Horan (1962 and 1963), Marter (1965), DOE (1979, 1980, 1981, 1982, and 1983), Hoff et al. (1984, 1985, 1986, 1987, 1989, 1990, and 1991), Chew and Mitchell (1988), and ERDA (1977). As with the releases discussed in the previous section, we evaluated these releases in a manner similar to that used for the Experimental Release Tests.

MTR Stack Release. On December 17, 1958, a rupture or fission break in the GEH-4 experimental capsule occurred, resulting in an estimated maximum release of 3000 Ci of krypton and xenon fission products (Sommers 1958b). Another similar fission break incident related to the GEH-4 experiment occurred on May 1, 1958 (Sommers 1958a), but release estimates were not made for this incident so we were unable to evaluate it. We evaluated the December release assuming a wind speed of 2 m s^{-1}, atmospheric stability class F, and a straight-line trajectory to the Frenchman's Cabin, the nearest offsite exposure location. We assumed the fission gas activity consisted entirely of ^{88}Kr. Based on these assumptions, the relative screening value for the onsite location was 1.8×10^{-7} and for the offsite location was 4.1×10^{-8}.

Blower Failure at ICPP. A blower failure at the ICPP in August 1958 resulted in an atmospheric release of approximately 10 µCi of long-lived fission product activity (ERDA 1977). We evaluated this release assuming a wind speed of 2 m s^{-1}, atmospheric stability class F, and a straight-line trajectory to Atomic City, the nearest offsite exposure location. We also assumed the release was comprised entirely of ^{90}Sr. Based on these assumptions, the relative screening value for the onsite location was 6.2×10^{-13} and for the offsite location was 9.7×10^{-12}.

MTR Stack Release. Between April 30 and May 6, 1960, a total of 6371 Ci of fresh fission product gases (reported to be isotopes of krypton and xenon) and 1600 Ci of ^{41}Ar were released from the MTR stack (Johnson 1960). The release was reported to result from a fission break in a capsule related to the GEH-14 experiment. We evaluated the release assuming a wind speed of 2 m s^{-1}, atmospheric stability class D, and a straight-line trajectory to the Frenchman's Cabin, the nearest offsite exposure location. We assumed the fission gas activity consisted entirely of ^{88}Kr. Based on these assumptions, the relative screening value for the onsite location was 7.2×10^{-8} and for the offsite location was 1.6×10^{-8}.

ETR Stack Release. Between June 20 and 21, 1960, ^{138}Cs and $^{88/89}Rb$ were released at a maximum rate of 130 Ci day^{-1} and an average rate of 85 Ci day^{-1} for a total release of 170 Ci over the 2-day period (Rich 1960). The release was reported to result from a fission break in a capsule related to the GEH-14 experiment. We evaluated the release assuming a wind speed of 2 m s^{-1}, atmospheric stability class F, and a straight-line trajectory to the Frenchman's Cabin, the nearest offsite exposure location. We assumed the released activity consisted entirely of ^{138}Cs, the

radionuclide with the largest screening factor. Based on these assumptions, the relative ranking value for the onsite location was 1.0×10^{-8} and for the offsite location was 3.9×10^{-9}.

MTR Stack Release. On February 22, 1963, a fission product activity release occurred at MTR as a result of a rupture in the NAA-74-1 lead experiment (Johnson 1963). The release via stack effluent was estimated to consist of 20,200 Ci of gaseous fission products. We evaluated the release assuming a wind speed of 2 m s^{-1}, atmospheric stability class F, and a straight-line trajectory to the Frenchman's Cabin, the nearest offsite exposure location. We assumed the fission gas activity consisted entirely of ^{88}Kr. Based on these assumptions, the relative screening value for the onsite location was 1.2×10^{-6} and for the offsite location was 2.8×10^{-7}.

WCF Ruthenium Releases. We located original data regarding daily ruthenium discharges from the WCF between December 1963 and October 1964 as well as percent composition of waste calciner off-gas for December and January (years unknown, assumed to be 1963 and 1964, respectively) during the course of Task Order 6 document reviews (Anonymous 1964). To evaluate the potential impacts of these releases, we selected the highest daily release (54.6 Ci on October 16, 1964) and assumed atmospheric stability class F, and we also selected the highest monthly release (96.8 Ci in October 1964) and assumed atmospheric stability class D. We used the reported percent composition of waste calciner off-gas to apportion the reported release between the isotopes that were likely to have been present (2.67% for ^{90}Sr, 93.14% for ^{106}Ru, 0.15% for ^{134}Cs, 2.89% for ^{137}Cs, and 1.15% for ^{144}Ce).

We evaluated the October 16 release assuming a wind speed of 2 m s^{-1}, atmospheric stability class F, and a straight-line trajectory to Atomic City, the nearest offsite exposure location. Based on these assumptions, the relative screening value for the onsite location was 1.2×10^{-6} and for the offsite location was 4.2×10^{-6}.

We evaluated the October monthly release assuming a wind speed of 2 m s^{-1}, atmospheric stability class D, and a straight-line trajectory to Atomic City, the nearest offsite exposure location. Based on these assumptions, the relative screening value for the onsite location was 3.6×10^{-7} and for the offsite location was 1.2×10^{-6}.

ICPP Iodine Release. A total of 1.48 Ci radioiodine was released from the ICPP on October 31, 1966 (Horan 1966). We evaluated this release assuming a wind speed of 2 m s^{-1}, atmospheric stability class F, and a straight-line trajectory to Atomic City, the nearest offsite exposure location. We assumed the release consisted entirely of ^{131}I. Based on these assumptions, the relative screening value for the onsite location was 3.9×10^{-9} and for the offsite location was 2.1×10^{-7}.

ECF Iodine Release. A suspected iodine release occurred at the ECF at the NRF on November 7, 1966 (O'Neill 1966). The magnitude of the release was not reported, but hand calculations made by Start (1966) indicate an upper bound release of 354 mCi. We evaluated this release assuming a wind speed of 2 m s^{-1}, atmospheric stability class F, and a straight-line trajectory to Atomic City, the nearest offsite exposure location. We assumed the release consisted entirely of ^{131}I. Based on these assumptions, the relative screening value for the onsite location was 9.4×10^{-10} and for the offsite location was 5.1×10^{-8}.

WCF Release. In May 1972, 20 Ci of long-lived particulate activity were released from the WCF (Anonymous, date missing). No additional data regarding the composition of this release were provided. Considering the relative lack of data regarding this release, duration of the release, the quantity of the release, and the probable composition of the release based on other WCF off-gas data, there are other releases with greater potential offsite exposure implications. This release was not evaluated further.

ICPP Release. On June 26, 1974, approximately 750 mCi of activity was released from the ICPP stack (Commander 1974). The cause of the release was not determined. The release consisted of 219 mCi ^{137}Cs, 219 mCi ^{90}Sr, 270 mCi ^{106}Ru/Rh, 21 mCi ^{125}Sb, 12 mCi ^{134}Cs, and 10 mCi ^{144}Ce/Pr. We evaluated this release assuming a wind speed of 2 m s^{-1}, atmospheric stability class F, and a straight-line trajectory to Atomic City, the nearest offsite exposure location. Based on these assumptions, the relative screening value for the onsite location was 2.0×10^{-8} and for the offsite location was 3.7×10^{-7}.

ICPP Releases. Releases of ^{85}Kr and ^{125}Sb are reported for the ICPP stack for August, September, and October 1974 (Keller 1974a; 1974b). From August 2 to 27, 1974, approximately 39,200 Ci and 0.570 Ci of ^{85}Kr and ^{125}Sb were released, respectively. From September 27 to October 27, 1974, approximately 55,750 Ci and 2.36 Ci of ^{85}Kr and ^{125}Sb were released, respectively. We evaluated the larger release during late September and most of October assuming a wind speed of 2 m s^{-1}, atmospheric stability class D, and a straight-line trajectory to Atomic City, the nearest offsite exposure location. Based on these assumptions, the relative screening value for the onsite location was 4.1×10^{-8} and for the offsite location was 3.9×10^{-8}.

1977 Unplanned Releases. Anderson (1978) discusses two unplanned airborne releases during 1977. On January 8, 1977, a release occurred from a charger load of EBR-II cut fuel assemblies. The estimated release from the cask consisted of 200 mCi ^{144}Ce/Pr, 20 mCi ^{95}Zr/Nb, 12 mCi ^{106}Ru/Rh, and 2 mCi ^{137}Cs. It was estimated that 20% of the activity escaped the building. We evaluated this release assuming a wind speed of 2 m s^{-1}, atmospheric stability class F, and a straight-line trajectory to Atomic City, the nearest offsite exposure location. We assumed the entire cask release escaped the building and was emitted to the atmosphere. Based on these assumptions, the relative screening value for the onsite location was 3.9×10^{-9} and for the offsite location was 6.9×10^{-9}.

On December 14, 1977, the solids transport line leading to the calcine waste storage vault (CPP-647) developed a leak (Anderson 1978). Radioactivity released to the atmosphere from this incident was estimated to be approximately 1 mCi, consisting of 72% ^{137}Cs, 25.2% ^{90}Sr, 2.5% ^{134}Cs, and 0.2% ^{154}Eu. We evaluated this release assuming a wind speed of 2 m s^{-1}, atmospheric stability class F, and a straight-line trajectory to Atomic City, the nearest offsite exposure location. Based on these assumptions, the relative screening value for the onsite location was 1.7×10^{-11} and for the offsite location was 7.0×10^{-10}.

Release due to APS Failure. A failure of the APS at the ICPP occurred on November 14, 1977, resulting in an estimated release of 67.7 mCi (Williamson 1977). The composition of the release was not reported. We evaluated this release assuming a wind speed of 2 m s^{-1}, atmospheric stability class F, and a straight-line trajectory to Atomic City, the nearest offsite

exposure location. We assumed the release consisted entirely of ^{90}Sr. Based on these assumptions, the relative screening value for the onsite location was 4.2×10^{-9} and for the offsite location was 6.6×10^{-8}.

ICPP Release. On February 22, 1981, an estimated 950 Ci of ^{85}Kr was released to the atmosphere via the ICPP stack as a result of pressure testing being conducted on a section of the Rare Gas Plant containing storage tanks WM-158A and WM-158B (Anonymous 1981). We evaluated this release assuming a wind speed of 2 m s^{-1}, atmospheric stability class F, and a straight-line trajectory to Atomic City, the nearest offsite exposure location. Based on these assumptions, the relative screening value for the onsite location was 4.2×10^{-9} and for the offsite location was 9.7×10^{-10}.

Calcine Bin Vent Filter Failure Release. On June 25, 1984, approximately 600 μCi of activity was estimated to be released to the environment from the 5th calcine bin set at the ICPP (Ikenberry 1984a; 1984b). The release was caused by failure of the off gas filters between the bin set cyclone cell and exhaust ventilation system and was estimated to consist of 98% ^{106}Ru and 2% ^{137}Cs. We evaluated this release assuming a wind speed of 2 m s^{-1}, atmospheric stability class F, and a straight-line trajectory to Atomic City, the nearest offsite exposure location. Based on these assumptions, the relative screening value for the onsite location was 1.3×10^{-11} and for the offsite location was 2.8×10^{-11}.

Comparison with Historical Dose Evaluation Data

To examine potential biases in our methodology, we evaluated relative screening values by pathway for the episodic releases evaluated as part of this work and compared them to dose values by pathway for a number of episodic releases evaluated by DOE (1991a), where available information enabled it. We discussed earlier the potential for biases created by the use of NCRP screening factors designed to evaluate long-term routine operations and noted the likely higher contribution by the ingestion pathway based on the use of these screening factors for episodic releases. In this section, we evaluated this potential bias for a number of different types of episodic releases, including longer and shorter term releases consisting of fresh fission products (e.g., IET runs and criticality/reactor accidents); shorter term releases consisting of aged fission products (e.g., FEBT and FPFRT releases); and releases consisting of single radionuclides (RaLa and CERT releases). Figures 22 through 28 compare dose and relative screening values by pathway for several episodic releases evaluated by DOE (1991a) and as part of this work.

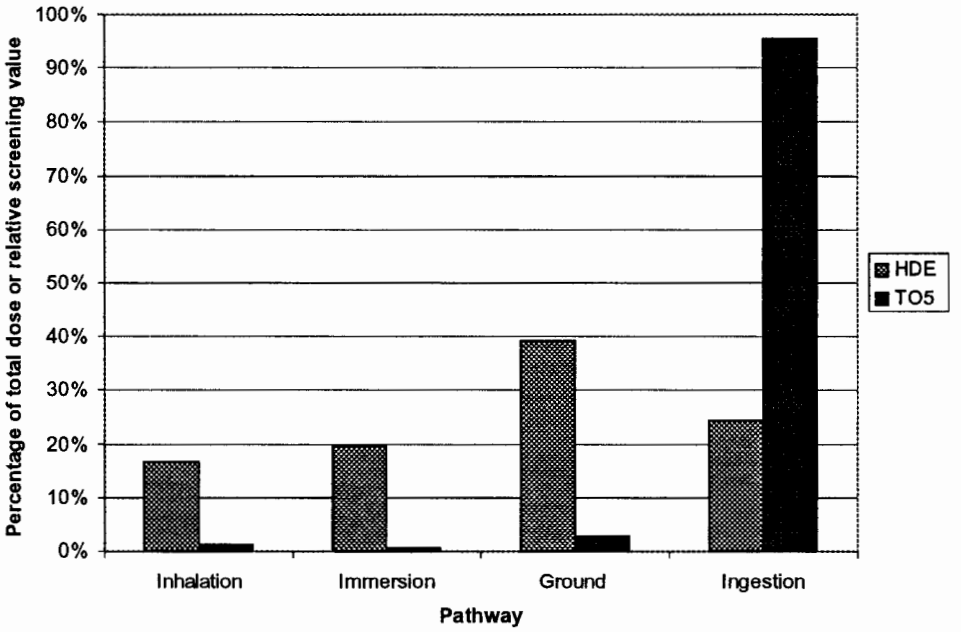

Figure 22. Percentage by pathway of total dose or relative screening value based on DOE (1991a) (HDE) and this work (TO5) for IET-3.

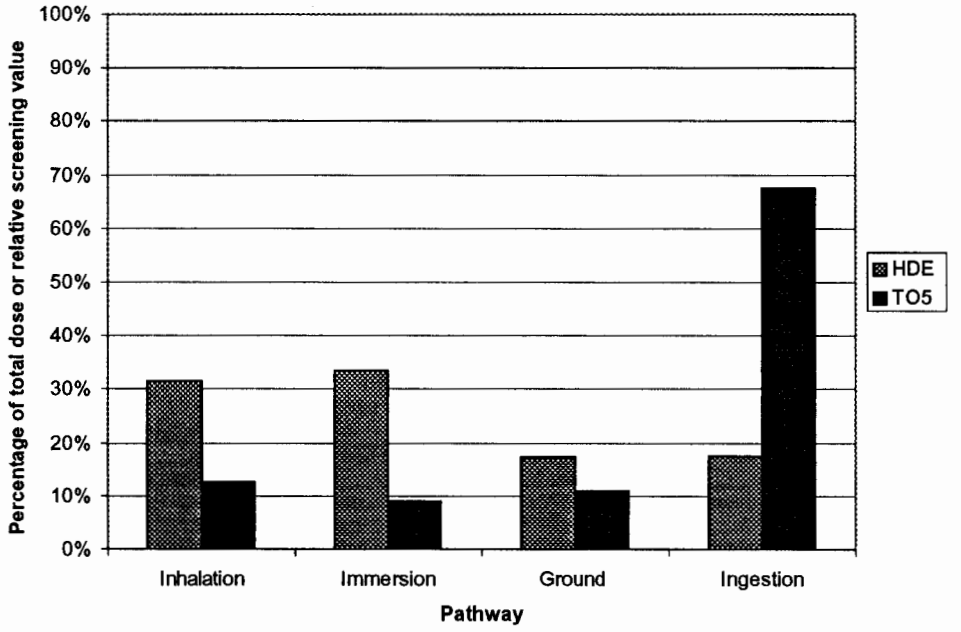

Figure 23. Percentage by pathway of total dose or relative screening value based on DOE (1991a) (HDE) and this work (TO5) for IET-13.

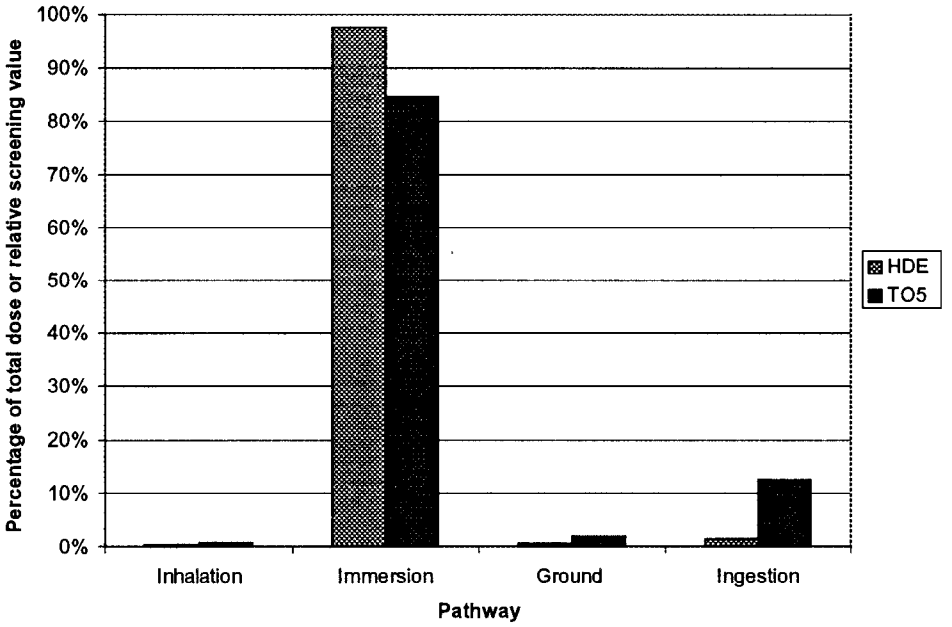

Figure 24. Percentage by pathway of total dose or relative screening value based on DOE (1991a) (HDE) and this work (TO5) for IET-16.

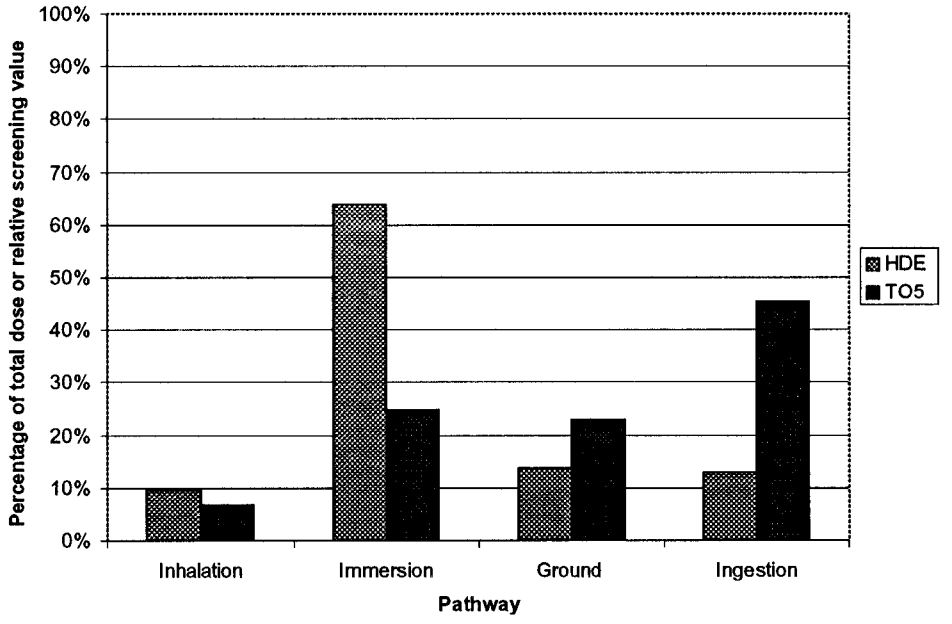

Figure 25. Percentage by pathway of total dose or relative screening value based on DOE (1991a) (HDE) and this work (TO5) for the ICPP 1959 criticality incident.

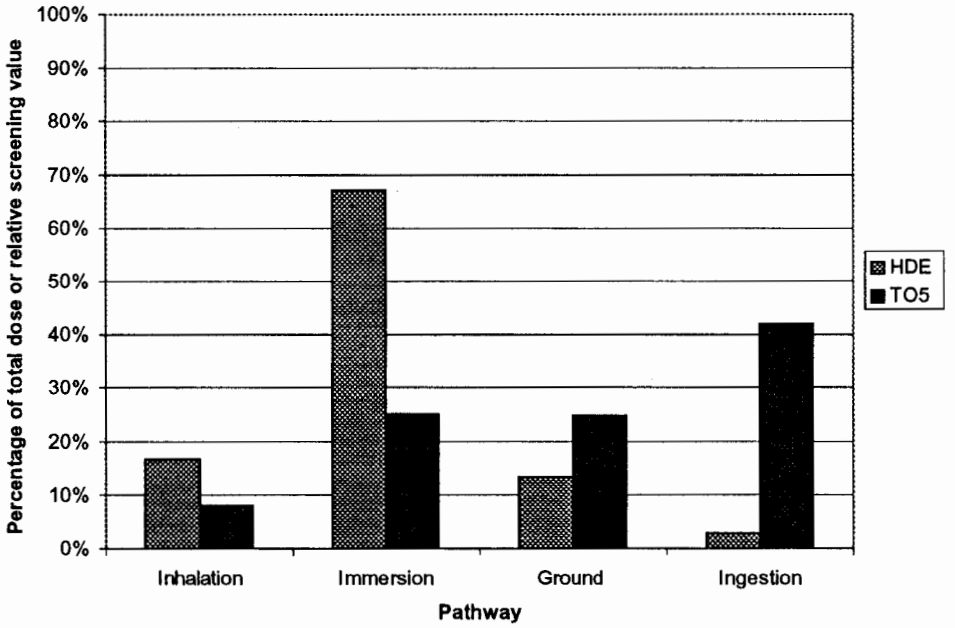

Figure 26. Percentage by pathway of total dose or relative screening value based on DOE (1991a) (HDE) and this work (TO5) for the SPERT-I test release.

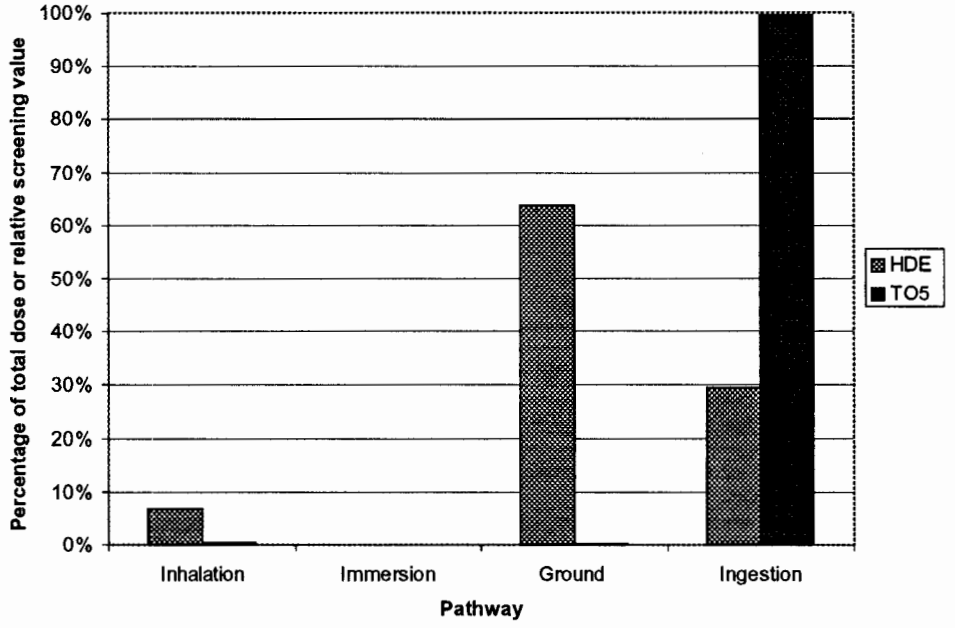

Figure 27. Percentage by pathway of total dose or relative screening value based on DOE (1991a) (HDE) and this work (TO5) for the FPFRT-1 test release.

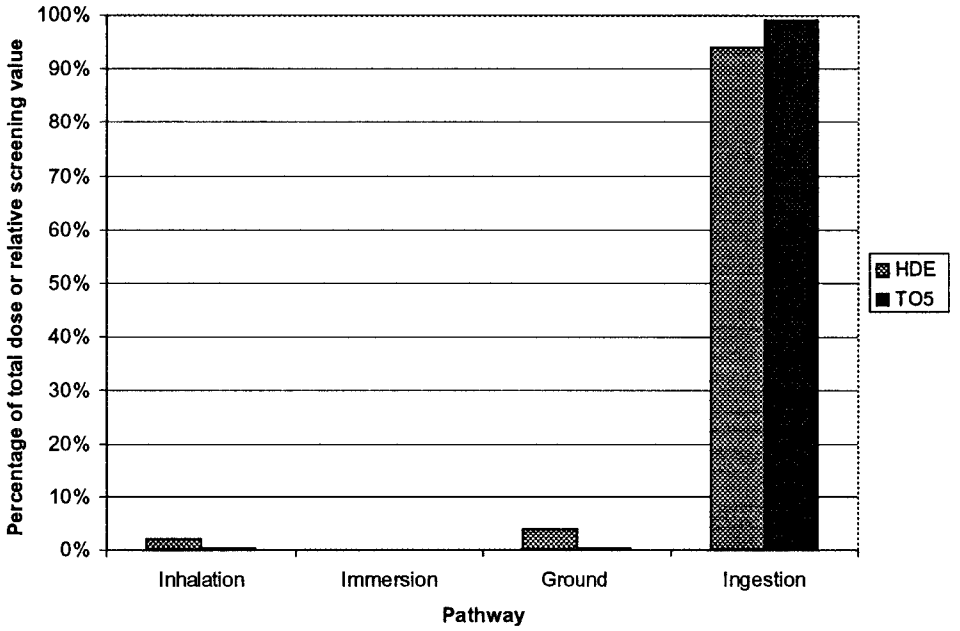

Figure 28. Percentage by pathway of total dose or relative screening value based on DOE (1991a) (HDE) and this work (TO5) for the CERT #1 test release.

These comparisons also illustrate the bias toward the ingestion pathway that using the NCRP screening factors introduces, which is evident in each example shown in Figures 22 through 28 We do not believe this precludes use of the NCRP screening factors for this purpose, but we do believe it is important to point out and examine potential biases introduced by using the screening factors for short term, episodic-type releases. We believe the screening factors represent the most practical option for evaluating the episodic releases because they allow for relatively simple evaluation by pathway of a given release. This would not be possible if actual dose conversion factors were used because specific pathway analysis would necessarily require the incorporation of a number of parameters, including deposition velocity, uptake rates into terrestrial food chains, and inhalation and consumption rates as well as more detailed atmospheric modeling methodology. While this would be important for more detailed dose assessments, it was not practical for a screening calculation to determine the potentially most important releases, which may require more detailed dose calculations.

A number of factors also influenced the relative importance of a given pathway for a particular release, including the assumed decay time (both during transit to the Site boundary and before the release); the assumed ^{41}Ar and uranium releases; the assumed release fractions; assumed reactor operating parameters; and the specific radionuclide constituents of the release.

The impact of a short decay time and a high assumed release fraction for noble gases, for example, can be seen by the relatively greater importance of the inhalation, plume immersion, and external exposure from ground contamination pathways for the 1959 ICPP criticality (Figure 25) and SPERT-I (Figure 26) releases. For the 1959 ICPP criticality, this is a result of the relatively short decay time before exposure as well as the large assumed release fraction for noble gases. This increases the importance of short-lived noble gas fission product particulate daughters, such as ^{138}Cs and ^{89}Rb, which deliver most of their potential dose via these three

pathways. Additionally, the particularly short decay time of 30 minutes for this release increased the relative importance of other short-lived noble gas fission product particulate daughters not considered by DOE (1991a), including ^{139}Cs and ^{93}Sr.

The impact of assumed ^{41}Ar and uranium release values as well as release fractions can be seen by comparing the IET- 3 (Figure 22), IET-13 (Figure 23), and IET-16 (Figure 24) releases. The ingestion pathway dominated for the IET-3 release because of the relatively large assumed halogen release fraction and the importance of the ingestion pathway for the halogen ^{131}I. The inhalation pathway was, by comparison, more important for the IET-13 release because of the magnitude of the assumed uranium releases relative to the fission product release. The immersion pathway dominated for the IET-16 release because of the magnitude of the assumed ^{41}Ar release by comparison to the relatively small fission product release.

The impact of a release consisting primarily of aged fission products can be seen by the domination of the ingestion pathway for the FPFRT-1 release (Figure 27). The longer-lived fission products ^{90}Sr and ^{137}Cs, in particular, were responsible for this as they delivered most of their potential dose via this pathway. The specific make-up of a given release was also an important factor in determining which pathway became the most important. For example, it is evident for the CERT #1 release (Figure 28) that the ingestion pathway was most important. This was because the release consisted entirely of ^{131}I, which delivered most of its potential dose via this pathway. Similar ingestion pathway importance would be expected for other releases involving primarily ^{131}I, such as other CERT and RDT test releases as well as RaLa releases.

We also examined the importance of including the additional radionuclides selected for this analysis by comparison to the list of radionuclides used in the DOE (1991a) analyses. This was done for those releases requiring extensive source term reconstruction and involving many fission products. We evaluated the relative screening values for offsite exposure locations and assessed the percent difference between the values calculated using the radionuclides selected for this work compared to the radionuclides selected by DOE (1991a).

Table 21 shows the three releases potentially impacted most by inclusion of the additional radionuclides selected for this work. The remaining releases all had values calculated using the DOE (1991a) subset of radionuclides within 5% of the values calculated using the subset of radionuclides selected for Task Order 5. The release events shown in Table 21 were all impacted by relatively short transit times to the nearest exposure location, which increased the potential importance of short-lived radionuclides, many of which were not evaluated by DOE (1991a). In particular, the differences were related to our inclusion of a number of radionuclides without existing screening factors, including ^{146}Ce, ^{139}Cs, and ^{93}Sr. This suggests that potential exposure to these radionuclides may warrant further investigation, particularly for the 1959 ICPP criticality event. It should be noted, however, that the relative importance suggested by this screening analysis for many of the shorter-lived radionuclides may be exaggerated because of the assumed annual average air concentration.

Table 21. Percent Difference between Relative Screening Values

Release	Relative screening values		Percent difference
	Task Order 5[a]	DOE (1991a)[b]	
1959 ICPP Criticality	4.48×10^{-7}	3.38×10^{-7}	24.5
SNAPTRAN-3	8.54×10^{-11}	7.13×10^{-11}	16.6
BORAX-I	1.63×10^{-9}	1.52×10^{-9}	6.7

[a] Based on important radionuclides selected as part of this work

[b] Based on important radionuclides selected by DOE (1991a)

Discussion of Results of Screening Episodic Events

We evaluated a total of 134 individual release events. By comparison, DOE (1991a) evaluated and calculated doses for a total of 54 episodic release events, including several IET runs that were evaluated during more than one time period. The simplified screening methodology we employed allowed a more comprehensive approach to evaluating each episodic release event for which it was possible to construct a source term. Additionally, our searches and reviews of the Task Order 6 enabled us to identify a number of release events not addressed by DOE (1991a). We attempted to evaluate all of the episodic releases in a consistent, conservative manner. Where specific information was unavailable for a given release, we made conservative assumptions.

We carefully reviewed the assumptions made as part of the analysis carried out by DOE (1991a) and generally believe that the work is thorough and scientifically defensible. However, we felt there were several opportunities to assess issues that were not addressed by DOE (1991a). These included a number of releases that were not evaluated individually as episodic releases, the potential importance of short-lived radionuclides without NCRP screening factors, and the potential importance of possible onsite exposure scenarios.

To prioritize the releases evaluated as part of this work, we compiled our calculated relative screening values for each episodic release in an Excel spreadsheet. Also included in this file were the major assumptions made for each release event, including atmospheric stability class, wind speed, transit time, and release fractions. Because of the long duration and discontinuous nature of many of the release events, we divided the releases into four separate exposure categories, including

- single-day release, onsite exposure
- single-day release, offsite exposure
- multi-day release, onsite exposure
- multi-day release, offsite exposure.

We then sorted each group of relative screening values from highest to lowest, using the highest screening values in cases where more than one value was calculated for a given release (i.e., when existing meteorological conditions and transit times reported by DOE resulted in different possible wind speeds). Table 22 shows the releases with the highest relative screening values in each category.

Table 22. Highest Relative Screening Values for Episodic Releases

Duration	Location	Release	RRV[a]	Start date	Stop date
Single-Day	Onsite	1959 ICPP criticality[b]	2.4E-06	10/16/59	
		WCF ruthenium release	1.2E-06	10/16/64	
		MTR stack release	1.2E-06	2/22/63	
		FECF Filter Break[b]	6.4E-07	10/30/58	
		FEBT-B[b]	2.2E-07	3/20/57	
		MTR stack release	1.8E-07	12/17/58	
		RaLa daily release	1.3E-07	5/28/58	
		RaLa daily release	1.3E-07	3/1/58	
		RaLa daily release	1.1E-07	10/7/57	
		RaLa daily release	1.0E-07	5/13/58	
Single-Day	Offsite	FEBT-B[b]	1.8E-05	3/20/57	
		RaLa daily release	7.1E-06	5/28/58	
		RaLa daily release	6.9E-06	3/1/58	
		RaLa daily release	5.8E-06	10/7/57	
		RaLa daily releases[c]	4.5 to 5.6E-06	5/13/58	
		WCF ruthenium release	4.2E-06	10/16/64	
		FECF Filter Break[b]	4.0E-06	10/30/58	
Multi-Day	Onsite	SL-1 accident[b]	1.3E-05	1/3/61	d
		IET-10[b]	1.1E-05	12/20/57	3/6/58
		ICPP Pu release	1.3E-06	7/9/59	7/11/59
		IET-4[b]	9.6E-07	4/17/56	6/29/56
		WCF ruthenium releases	3.6E-07	10/1/64	12/31/64
		IET-22[b]	1.8E-07	8/12/60	8/25/60
		IET-24[b]	1.6E-07	10/17/60	10/26/60
		RaLa Run #2	1.6E-07	2/24/57	3/30/57
		IET-3[b]	1.5E-07	1/17/56	2/24/56
		RaLa Run #8	1.4E-07	10/21/57	1/5/58
Multi-Day	Offsite	IET-10[b]	4.0E-04	12/20/57	3/6/58
		IET-4[b]	3.8E-05	4/17/56	6/29/56
		SL-1 accident[b]	3.1E-05	1/3/61	d
		RaLa Run #2	7.8E-06	2/24/57	3/30/57
		RaLa Run #8	6.9E-06	10/21/57	1/5/58
		IET-6[b]	6.7E-06	9/24/56	1/3/57
		RaLa Run #1	5.6E-06	2/3/57	2/23/57
		IET-3[b]	5.6E-06	1/17/56	2/24/56
		IET-14[b]	5.4E-06	4/17/59	5/20/59
		IET-18[b]	5.4E-06	12/23/59	2/8/60

[a] Relative screening value

[b] Evaluated by DOE (1991a)

[c] Includes six separate RaLa daily releases with RRVs between 4.5×10^{-6} and 5.6×10^{-6}

[d] Incident occurred on a single day, but ^{131}I releases continued for several weeks

We highlighted those releases we believe may warrant further investigation and detailed calculation of doses. Our selections were based primarily on the relative screening values, but we considered other factors as well. Several of the releases have been previously evaluated by DOE (1991a), but they are potentially important for both onsite and offsite exposure scenarios. The potential onsite exposure importance as well as the uncertainty associated with assumed release fractions for IET-10, IET-4, SL-1, FEBT-B, and the FECF filter break may justify a more detailed evaluation of these release events. The potential exposures associated with the 1959 ICPP criticality and the SL-1 accident (particularly at onsite locations) may deserve a closer look into the importance of short-lived radionuclides without NCRP screening factors. The fact that the MTR stack release (February 22, 1963), the WCF ruthenium release (October 16, 1964), the ICPP plutonium release (July 9-11, 1959), and the RaLa runs and higher daily releases have not been evaluated previously as episodic releases also may also justify a more detailed assessment.

Combined, the events we selected are representative of the wide array of episodic-type releases that have occurred historically at the INEEL. More detailed dose evaluations for some number of the top releases in each category should enable a decision regarding the need for additional analyses for other, lower ranking releases. Several releases have high relative screening values in both onsite and offsite categories (e.g., FEBT-B, SL-1, WCF ruthenium releases, IET-10, IET-4, FECF filter break, 1959 ICPP criticality, and several RaLa releases), so a detailed analysis of one of these events would allow for an assessment of its overall importance in more than one category. The CDC, the INEEL HES, and other involved stakeholders should continue to work closely to determine which release events deserve further investigation into health impacts to potentially exposed members of the public, both at onsite and offsite locations.

DUCK HUNTER SCENARIO FOR THE INEEL

Our screening process was applied to three areas: (1) routine releases from the INEEL, (2) releases to groundwater and possible exposure through ingestion, and (3) releases from episodic events or accidents. Yet we may not have encompassed all potential exposure pathways at the INEEL. A member of the public may have been exposed to releases from the INEEL by exposure to ducks that may have resided on liquid waste ponds at the INEEL. Studies at the radioactive waste disposal ponds at the INEEL have shown the presence of radionuclides in wild waterfowl tissue samples (Halford et al. 1981; Markham et al. 1988). Birds are the most mobile of hunted species because of their migratory patterns and they could potentially move radionuclides from contamination sites into surrounding areas, where hunters could shoot and eat them. This section describes the results of a dose evaluation of a scenario of a pregnant female duck hunter. This scenario was developed with the help of the INEEL HES at the September 1999 quarterly meeting. Because this is a unique and potential pathway of concern for those living in the region around the INEEL, it is important to consider the results and implications of this scenario for the Task Order 5 radionuclide screening work.

The scenario involves a female hunter during her first trimester who shoots 180 ducks over the course of the duck-hunting season of three months from October through December. With a bag limit of 6 per day, this amounts to a total of 180 ducks. It was agreed that we are calculating dose to the person, not to the duck. Although she most likely would be hunting along the Snake River or in a wildlife management area away from the INEEL, we have assumed that the contaminated ducks have come from the TRA ponds with no loss of activity from the levels

measured in the 1980s. The TRA ponds have been used for disposal of low-level liquid radioactive wastes from the three experimental reactors at TRA. The ponds encompass an area of about 3 hectare (7.5 acres or 30,000 m^2). In reality, the closest lake to the site that is open to the public is a wildlife management area about 68 km from TRA ponds. From studies conducted at the INEEL, it is estimated that about a million ducks fly through the area in a season, and about 3000 stop on the TRA Ponds (Halford et al. 1981). The length of time waterfowl stay on the ponds is usually no more than 24 hours, although a few have stayed for a week or more. These screening dose calculations are based on measured radionuclide concentrations in tissues from waterfowl from the TRA ponds (Halford et al. 1981; Markham et al. 1988). Of the180 ducks shot, 12 are assumed to be contaminated with radionuclides at levels that have been measured and reported in these studies

Some exposure parameters and duck hunter behaviors were established through interaction with the INEEL HES members. The dead ducks are held against her abdomen for 5 minutes and then tossed in the back of a blind or the bottom of a boat. If the hunter was walking, the ducks were put in a bag and carried to the car and put in the trunk or back of a truck. Based on information from hunters in the group, ducks are usually "field dressed," i.e., gutted, but the feathers and skin remain. It was agreed that the liver is saved to eat later. Once at home, the hunter removes the duck feathers in the backyard for making a pillow. To ensure conservatism, the feathers are not washed or sorted but used directly to make a pillow. She then enters the house, removes her boots, and sits in an easy chair in her living room with hunting clothes on for 30 minutes. This sitting resulted in the transfer of dirt, residue, and contamination from the clothing falling onto the chair. We assumed the chair was not cleaned.

We assumed she cooks and eats the muscle and livers from all 12 contaminated ducks within a short period of time. Most people consume only the breast muscle of ducks but we assumed our hunter ate all the muscle and liver and we calculated an ingestion dose based on these assumptions. Ingestion dose calculations were reported here for ^{137}Cs, ^{134}Cs, ^{75}Se, ^{131}I, and $^{239,240}Pu$ contamination of muscle, and liver using average and maximum concentrations reported previously (Halford et al. 1981; Markham et al. 1988). The first three radionuclides had the highest concentrations in muscle tissue. In addition, we calculated doses from exposure to external irradiation from ^{137}Cs contamination on the chair, car seat, and in the feather pillow, and from inhalation of ^{137}Cs and $^{239,240}Pu$ contamination while constructing the pillow. We used average and maximum concentrations of ^{137}Cs and $^{239,240}Pu$ on feathers reported previously.

Table 23 lists the radionuclide concentrations measured previously in waterfowl taken from the TRA ponds. Our primary focus has been to assess the doses from ^{137}Cs because this was the focus of interest from the INEEL HES. We also calculated doses from $^{239,240}Pu$. The dose calculations for $^{239,240}Pu$ are more conservative because the concentrations used for the dose calculations were measured in duck tissues after the ducks were held in wire-enclosed cages on the TRA pond for 6 weeks to 5 months. As a result, we would expect the concentrations of $^{239,240}Pu$ to be higher than levels measured in ducks using TRA ponds as a resting area for a short time period on their migratory route.

Table 23. Reported Concentrations (pCi g^{-1}) of Radionuclides in Fresh Weight in Water Fowl from TRA Waste Ponds

	^{137}Cs[a]		$^{239,240}Pu$[b]		^{134}Cs[a]		^{75}Se[a]		^{131}I[a]	
Organ	Average	Max.	Average	Max.	Average	Max.	Average	Max.	Average	Max.
Muscle	732	4070	0.002[c]	0.0046	152	920	52	376	122	290
Liver	1060	3880	0.08	0.20	214	860	229	590	66	690
Feather	64	260	0.14	0.32						

[a] From Halford et al. (1981); concentrations measured in ducks using the ponds as a resting area, usually staying less than 24 hours.

[b] From Markham et al. (1988); ducks were held in a wire fence enclosure on the TRA ponds for 43-145 days.

[c] Concentration reported as *below detectable concentration*; we used the minimum detectable concentration as the average concentration for these calculations.

For ^{137}Cs, a beta and gamma emitter, we evaluated three potential exposure pathways for both the average and maximum concentrations and calculated doses from:

1. Ingestion of the muscle and liver of the 12 contaminated ducks. We assumed all contaminated ducks are eaten within a short time.
2. Exposure from external irradiation from contaminated feathers
    - Left on car seat and in an easy chair in the house assuming person sits in car for 2 hours per day and sits in easy chair for 3 hours per day for a year
    - Used in making a pillow and having contact with the pillow for 8 hours per day for a year.

    External irradiation is a potential pathway because ^{137}Cs decays by beta emission to the metastable ^{137m}Ba (half-life = 2.6 minutes) which in turn decays by emitting a 0.66 MeV gamma.
3. Inhalation of ^{137}Cs from radioactivity resuspended from the feathers to air in plucking feathers and from the pillow.

For $^{239,240}Pu$, an alpha emitter, we evaluated two potential exposure pathways for both the average and maximum concentrations and calculated doses from

1. Ingestion of $^{239,240}Pu$ from the muscle and liver of the 12 contaminated ducks. We assumed all contaminated ducks are eaten within a short time.
2. Inhalation of $^{239,240}Pu$ from radioactivity resuspended from the feathers to air in plucking feathers and from pillow.

For ^{134}Cs, ^{75}Se, and ^{131}I, we evaluated the ingestion pathway for both the average and maximum concentrations in muscle and liver. Table 24 summarizes the annual doses calculated for ^{137}Cs and $^{239,240}Pu$. The sections following the table carefully describe the assumptions made, methods used, and calculations of the doses from each pathway.

Table 24. Screening Doses from ^{137}Cs and $^{239,240}Pu$ from Potential Exposure Pathways to Duck Hunter

	Screening dose (mrem) from ^{137}Cs		Screening dose (mrem) from $^{239,240}Pu$	
Pathway	Average concentration [a]	Maximum concentration[a]	Average concentration [b]	Maximum concentration[b]
Ingestion of				
muscle from 12 ducks	140	760	0.007	0.017
liver from 12 ducks	15	56	0.022	0.055
muscle from 1 duck	12	64	0.006	0.0014
liver from 1 duck	1.3	4.7	0.0018	0.0046
Feathers – external exposure				
from pillow (8 h d^{-1})	0.58	2.3	na [c]	na
from car seat (2 h d^{-1})	0.07	0.30	na [c]	na
from easy chair (3 h d^{-1})	0.11	0.47	na [c]	na
Feathers – inhalation of resuspended material	0.13	0.56	0.002 (car) 0.0008 (chair)	0.008 (car) 0.003 (chair)

[a] From Halford et al. (1981).

[a] From Markham et al. (1988).

[c] na = not applicable; $^{239,240}Pu$ is an alpha emitter with no energetic gammas.

Calculation Details

The measured ^{137}Cs and $^{239,240}Pu$ concentrations in various tissues of ducks collected from the TRA ponds at the INEEL are given in Table 23. The concentrations are used as the basis for estimating the radiation dose from ingestion of muscle and liver from ducks, from inhalation, and for ^{137}Cs, direct gamma irradiation.

Ingestion Pathway

The basic formula for intake from the ingestion of duck meat is:

$$I_{meat} = C_{meat}(U_{meat})f_c(n) \quad (11)$$

where,

I = intake of radionuclide due to meat ingestion (pCi)
C_{tissue} = concentration of radionuclide in muscle and liver tissue (pCi g^{-1})
U_{duck} = amount of meat consumed per duck
f_c = fraction of duck meat consumed that is contaminated (dimensionless, 1.0)
n = number of ducks consumed.

The committed effective dose equivalent from ingesting the contaminated duck meat (muscle and/or liver) was then determined with the equation:

$$D_i = I_{duck\ tissue}\ (DCF_i) \quad (12)$$

where,

D_i = committed radiation dose equivalent due to ingestion of radionuclide *i* in meat from ducks residing on the TRA ponds (Sv y^{-1})

DCF_i = dose conversion factor for ingestion of radionuclide *i*.

For ^{137}Cs, the dose conversion factors for ingestion 1.3×10^{-8} Sv Bq^{-1} (4.8×10^{-5} mrem pCi^{-1}) and for inhalation (with slow clearance from the lung) 3.9×10^{-8} Sv Bq^{-1} (1.44×10^{-4} mrem pCi^{-1}) are used. For $^{239,240}Pu$, the dose conversion factors (DCF) for ingestion 2.5×10^{-7} Sv Bq^{-1} (9.2×10^{-4} mrem pCi^{-1}) and for inhalation (assuming slow clearance from the lung), 1.6×10^{-5} Sv Bq^{-1} (5.9×10^{-2} mrem pCi^{-1}) are used (ICRP 1995). Dose conversion factors for 134Cs, 75Se, and 131I are taken from the same source. The results of these calculations are shown in Table 25 for ^{137}Cs. ^{134}Cs, ^{75}Se, ^{131}I, and $^{239,240}Pu$. The dose from eating muscle from 12 ducks contaminated with the average ^{137}Cs concentration is 140 mrem y^{-1}. The dose the hunter would receive from eating all 12 livers from the contaminated ducks is about 10 times lower at 15 mrem y^{-1}. If we assume the maximum concentration of ^{137}Cs in all 12 ducks, then the annual dose would be 760 mrem from the muscle and 56 mrem from eating the liver.

In addition, to dose from ingesting muscle and liver from the contaminated ducks, we estimated doses to the hunter from the ^{137}Cs on the feathers from the contaminated ducks collected at the TRA ponds.

Inhalation Pathway

For the potential exposure pathways of inhalation and direct irradiation from the ^{137}Cs surface contamination of feathers, we determined the concentration of ^{137}Cs:

- Deposited on the seat of the car—We assumed that the hunter placed her 12 contaminated ducks on the car seat next to her and that 10% of the contamination on the feathers ws deposited on the car seat. She was then exposed to that external irradiation for 2 hours per day for 365 days for the year. We assumed an exposure area of 0.5 m (20 in) in diameter.
- Deposited on the easy chair at home—We assumed that the hunter returned from hunting, removed her boots, then sat in her easy chair at home for 30 minutes before removing her hunting jacket and pants. We assumed that 10% of the contamination from the feathers from the 12 contaminated ducks was deposited on her hunting clothes and that all of the contamination from the clothes was then deposited on the easy chair. We assumed she sat in her easy chair for 3 hours per day for 365 days and was exposed to the contamination in the easy chair. We assumed the exposure area of the chair was 1 square meter.
- In feathers used to make a pillow—We learned that geese are usually used as the source for pillows because geese have more down, with about 16 geese needed to make a standard sized pillow. It was estimated that it would take 70 ducks to make a down pillow using ducks; if all feathers were used in the pillow it would take fewer ducks. For our cautious calculation for this scenario, we assumed that all feathers, not just the down, were used in the pillow from

the 12 contaminated ducks. We assumed that 420 g (14 oz.) of feathers were used in making a round pillow with dimensions of 50 centimeters in diameter by 15 centimeters thick (1.6 feet diameter by × 0.5 feet thick). With these dimensions, the pillow had a surface area of 1960 cm^2 or 0.2 m^2 ($2ft^2$) and a volume of 29,000 cm^3 or 0.029 m^3 (1 ft^3). Thus, each of the 70 ducks contributed about 6 grams of feathers, and the 12 contaminated ducks contributed 72 grams of the 420 grams of feathers. We assumed that the remaining 80% of the contamination on the feathers from the 12 contaminated ducks (subtracting 10% deposited on the car seat and the 10% deposited on the hunter's clothes and eventually onto the easy chair) stayed on the feathers used to make the pillow and that the feathers were not washed before making the pillow.

Table 25. Screening Doses from Ingestion of Duck Meat from the TRA Ponds

		Screening dose (mrem)			
		Muscle[a]		Liver[b]	
Radionuclide	Scenario[c]	1 duck	12 ducks[c]	1 duck	12 ducks[d]
^{137}Cs	Average[d]	12	140	1.3	15
^{137}Cs	Maximum[d]	64	770	4.7	57
^{134}Cs	Average	3.6	43	0.4	5
^{134}Cs	Maximum	21	260	1.6	19
^{75}Se	Average	0.1	2	0.05	0.5
^{75}Se	Maximum	1	12	0.1	1.4
^{131}I	Average	0.4	5	0.1	2
^{131}I	Maximum	8.3	100	1.5	18
$^{239,240}Pu$	Average[e]	0.0006	0.007	0.002	0.022
$^{239,240}Pu$	Maximum[e]	0.0014	0.017	0.005	0.055

[a] Assume weight of live mallard of 1300 g (Halford et al. 1983); 25% of weight assumed to be muscle tissue (Johnson 1980; Halford et al. 1981).

[b] Duck liver weight is 25 g (Halford et al. 1981).

[c] For the dose calculations, we used the average and maximum concentrations (pCi g^{-1}) measured in fresh weight water fowl from the TRA leaching ponds (see Table 23).

[d] Bag limit was 6 ducks per day in Idaho; during the scenario development at the September 1999 INEEL HES meeting, the subcommittee agreed on the assumption that the hunter reached her limit every day for 3 months, resulting in 180 ducks over the 3-month hunting season. We further assumed 12 of the ducks were contaminated. We also assumed the ducks come from the TRA ponds.

[e] Measured concentrations of $^{239,240}Pu$ in waterfowl from the TRA ponds (from Markham et al. 1988).

For the inhalation pathway, we assumed the loose surface contamination could be resuspended in the air and be available for inhalation. The degree of hazard from surface contamination was strongly dependent on the degree to which the contaminant was fixed to the surface. For inhalation, the relationship between the concentration of loose surface contamination on the feathers and the concentration in air above the contaminated surface can be defined by the resuspension factor, f_r, and is the concentration in air divided by the surface concentration. It is defined by:

$$f_r = C_a \div C_s \text{ (concentration in volume of pillow (Bq m}^{-3}\text{)} \quad (13)$$

where,

f_r = resuspension factor (m^{-1})
C_a = air concentration (Bq m^{-3})
C_s = concentration on surface (Bq m^{-2})

And to calculate the concentration in air:

$$C_a = f_r \times C_s \quad (14)$$

Measured values of resuspension (f_r) of loose surface contamination shows that the resuspension factor varies from about 10^{-4} to 10^{-8} m^{-1} (Cember 1988). For our calculation we will assume a value of 10^{-5} m^{-1}. For the radionuclides on the car seat, we assumed that 10% of the measured concentrations were deposited on the car seat. For inhalation, the concentration on the surface of the car seat (0.5 m diameter area) was calculated for ^{137}Cs as:

$$C_s \text{ for } ^{137}Cs = (6.4 \text{ pCi g}^{-1} \times 72 \text{ g feathers})/0.2 \text{ m}^2$$
$$C_s \text{ for } ^{137}Cs = 2304 \text{ pCi m}^{-2}$$

Assuming a resuspension factor of 10^{-5} m^{-1}, the corresponding ^{137}Cs air concentration would be 2.3×10^{-2} pCi m^{-3}. For inhalation, the concentration on the surface of the *easy chair* (1.0 m diameter area) was calculated for ^{137}Cs as:

$$C_s \text{ for } ^{137}Cs = (6.4 \text{ pCi g}^{-1} \times 72 \text{ g feathers})/0.8 \text{ m}^2$$
$$C_s \text{ for } ^{137}Cs = 580 \text{ pCi m}^{-2}$$

Assuming a resuspension factor of 10^{-5} m^{-1}, the ^{137}Cs air concentration near the chair would be 5.8×10^{-3} pCi m^{-3}. Similar calculations for $^{239,240}Pu$ yielded surface concentrations from the car seat of 5 pCi m^{-2} and from the easy chair of 1.3 pCi m^{-2}. Assuming a resuspension factor of 10^{-5} m^{-1}, the corresponding $^{239,240}Pu$ air concentration above the car seat would be 5×10^{-5} pCi $m^{-3}$and for the easy chair was 1.3×10^{-5} pCi m^{-3}.

For the *pillow,* we calculated the concentrations of ^{137}Cs and $^{239,240}Pu$ in the pillow based on the average measured concentration of ^{137}Cs in feathers of 64 pCi g^{-1} (2.4 Bq g^{-1}) and of $^{239,240}Pu$ in feathers of 0.14 pCi g^{-1}. We assumed 6 grams of feathers per duck for our 12 contaminated ducks and calculated the concentration of ^{137}Cs and $^{239,240}Pu$ in our pillow (volume of 0.029 m^3). We assumed that 80% of the concentration remained on the feathers used for the pillow.

For the average measured concentration of ^{137}Cs in the feathers, the concentration on the surface of the pillow assuming all contamination is on the surface of the pillow would be

$$C_s = (0.8 \times 64 \text{ pCi g}^{-1} \times 72 \text{ g/pillow})/0.029 \text{ m}^3 = 1.3 \times 10^5 \text{ pCi m}^{-3} (1.3 \times 10^{-7} \text{ Ci m}^{-3}).$$

Then the atmospheric concentration (pCi m^{-3}) = 10^{-5} (1.3×10^5 pCi m^{-3}) = 1.3 pCi m^{-3}.

For the maximum concentration of ^{137}Cs measured in feathers of 260 pCi g^{-1} (assuming that 80% of the contamination on the feathers remains), the concentration on the surface of the pillow assuming all contamination is on the surface of the pillow was

$$C_s = (0.8 \times 260 \text{ pCi g}^{-1} \times 72 \text{ g/pillow})/0.029 \text{ m}^3 = 5.2 \times 10^5 \text{ pCi m}^{-3} (5.2 \times 10^{-7} \text{ Ci m}^{-3})$$

and, the atmospheric concentration was (pCi m^{-3}) = 10^{-5} (5.2×10^5 pCi m^{-3}) = 5.2 pCi m^{-3}.

Tables 26 and 27 summarize these values and show how they were used to calculate the dose from inhalation of air with these concentrations of ^{137}Cs and $^{239,240}Pu$.

Table 26. Screening Doses from Inhalation of ^{137}Cs Contamination on Duck Feathers

Source of contaminated air	Air Concentration (pCi m^{-3})	Breathing rate (m^3 h^{-1})[a]	Contact time (hr per y)	Breathing volume (m^3 y^{-1})	Inhalation dose coefficient (mrem pCi^{-1})	Dose from inhalation (mrem)
Car Seat						
Average ^{137}Cs concentration[b]	2.30×10^{-2}	0.9	730	660	1.4×10^{-4}	0.002
Maximum ^{137}Cs concentration[b]	9.2×10^{-2}	0.9	730	660	1.4×10^{-4}	0.0085
Easy Chair						
Average ^{137}Cs concentration[b]	5.8×10^{-3}	0.9	1095	985	1.4×10^{-4}	0.0008
Maximum ^{137}Cs concentration	2.3×10^{-2}	0.9	1095	985	1.4×10^{-4}	0.0032
Pillow						
Average ^{137}Cs concentrationa	1.3	0.9	2920	2630	1.4×10^{-4}	0.48
Maximum ^{137}Cs concentration	5.2	0.9	2920	2630	1.4×10^{-4}	1.9

[a] Based on conservative breathing rate estimate of 8000 m^3 y^{-1} (NCRP 1996).
[b] Sum of air concentrations from contamination in pillow, on car seat, and on easy chair.

Table 27. Screening Doses from Inhalation of ^{239,240}Pu Contamination on Duck Feathers

Source of contaminated air	Air Concentration (pCi m^{-3})	Breathing rate (m^3 h^{-1})[a]	Contact time (hr per y)	Breathing volume (m^3 y^{-1})	Inhalation dose coefficient (mrem pCi^{-1})	Dose from inhalation (mrem)
Car Seat						
Average ^{239,240}Pu concentration[b]	5.0×10^{-5}	0.9	730	660	5.9×10^{-2}	0.002
Maximum ^{239,240}Pu concentration[b]	2.0×10^{-4}	0.9	730	660	5.9×10^{-2}	0.008
Easy Chair						
Average ^{239,240}Pu concentration[b]	1.3×10^{-5}	0.9	1095	985	5.9×10^{-2}	0.0008
Maximum ^{239,240}Pu concentration[b]	5.2×10^{-5}	0.9	1095	985	5.9×10^{-2}	0.003

[a] Based on conservative breathing rate estimate of 8000 m^3 y^{-1} (NCRP 1996).
[b] Sum of air concentrations from contamination in pillow, on car seat, and on easy chair.

External Irradiation Pathway

For the contribution to dose from external irradiation from the feathers in the pillow and from holding the duck near the body after hunting and gutting, we used the quantitative relationship between dose rate and distance from a volume radiation source. Cesium-137 decays by beta emission to the metastable ^{137m}Ba (half-life = 2.6 min), which in turn decays by emitting a 0.66 MeV gamma. The radiation exposure from a volume containing uniformly distributed gamma emitting isotope was estimated from the effective surface activity after allowing for self absorption within the volume (Cember 1988), using the following equation:

$$d(C_a) = C_v \times dx \times e^{-\mu} \quad (15)$$

Integrating the Equation (15) over the total thickness, t, yielded the effective surface activity as:

$$C_a = C_v \times e^{-\mu x} \times dx = C_v/\mu\,(1-e^{-\mu t}) \quad (16)$$

where,

C_a = activity on surface due to radioactivity in volume (pCi m^{-2})
C_v = concentration of radionuclide in volume (pCi m^{-3})
μ = linear absorption coefficient of material (m^{-1}); for ^{137}Cs, we assumed the material is air with a linear absorption coefficient for 0.66 MeV gamma of 0.0035 m^{-1}
t = thickness of pillow, 0.15 m.

We used this concentration (C_a) to calculate the dose equivalent rate at a specified distance from the pillow using Equation (17):

$$H = \pi \times \Gamma \times (C_v/\mu)\,(1-e^{-\mu t})\ \ln[(R2 + h2)/h2] \quad (17)$$

where,

H = dose equivalent rate (rem h^{-1})
Γ = specific gamma ray emission; for ^{137}Cs $\Gamma = 2.3 \times 10^{-7}$ (X-m^2)/(MBq-hr) (Cember 1988)
R = distance from center to edge of source (m)
h = height above source; we choose a conservative value of 0.001 m.

For the average concentration of ^{137}Cs in the feathers, the dose equivalent rate was

$$H = \pi [2.3 \times 10^{-7} (X\text{-}m^2)/(MBq\text{-}hr)] \times (4.72 \times 10^{-3} MBq\ m^{-3}/0.0035\ m^{-1}) \times (1\text{-}e^{(-0.0035/m \bullet 0.15\ m)}) \times \ln [(0.25\ m)^2 + (0.001\ m)^2]/0.001\ m)^2.$$

$$H = 2 \times 10^{-9} Sv\ hr^{-1}, \text{ or } 2 \times 10^{-7} rem\ hr^{-1}.$$

This is equivalent to 2×10^{-4} mrem h^{-1} from external irradiation from the ^{137}Cs contaminated feathers in the pillow. Assuming contact with the pillow for 8 h d^{-1}, for 365 days, the dose equivalent is 0.58 mrem y^{-1}.For the maximum concentration of ^{137}Cs in the feathers, the dose equivalent rate would be 2.3 mrem y^{-1}.

Although the hunter was pregnant, no fetal doses were calculated. However, as a conservative approach, one could assume that the fetus received the same dose as the mother from the pathways evaluated. Ingestion of a contaminated duck with the maximum measured ^{137}Cs concentration resulted in a dose of 64 mrem (0.00064 Sv).

Duck Hunter Scenario Findings

Ingestion screening dose calculations are reported here for ^{137}Cs, ^{134}Cs, ^{75}Se, ^{131}I, and $^{239,240}Pu$ contamination of muscle and liver, using average and maximum concentrations reported previously. The first three radionuclides had the highest concentrations measured in muscle tissue. Ingesting the duck meat was the most important exposure pathway and ^{137}Cs was the largest contributor to the ingestion dose. These conservative calculations show that the screening dose from ingestion of duck meat contaminated with ^{137}Cs is the largest contributor to screening dose to the hypothetical female duck hunter. Based on these very conservative assumptions, we calculated a screening dose from eating one contaminated duck with the maximum concentration as 64 mrem (or 12 mrem if the average ^{137}Cs concentration was used in the calculations). If 12 contaminated ducks, with the average ^{137}Cs concentration in muscle, were eaten at one sitting, the hunter's screening dose would be approximately 140 mrem (0.0014 Sv). The hypothetical screening doses from external irradiation from the feathers under several different situations (contaminated car seat, easy chair, and pillow) and from the inhalation of resuspended materials from the feathers contributed less than 1% of the dose to the hunter. Screening doses to the hunter from $^{239,240}Pu$ contamination of duck meat and feathers were 3 to 4 orders of magnitude lower than screening doses from ^{137}Cs. It is important to note that these calculations are based on numerous conservative assumptions that tend to maximize the potential dose to the hypothetical person. If the screening dose is low under these very conservative conditions, then it is reasonable to assume that the doses under more realistic situations (e.g., eating meat from one contaminated duck in a year) would be considerably lower.

This screening dose evaluation suggests that eating meat from a contaminated duck that rested on the TRA ponds is a potential and realistic exposure pathway. While the ingestion pathway may be a potential pathway of exposure for those offsite, the other potential pathways of exposure from duck hunting are much less important.

SUMMARY AND CONCLUSIONS ON KEY RADIONUCLIDES, OPERATIONS AND EPISODIC RELASES BASED ON SCREENING METHODOLOGY

It is evident that the vast majority of both atmospheric and liquid routine releases have been from the ICPP and TRA facilities. Total discharges from ICPP, TRA, TAN, and ANL-W have accounted for greater than 95% of the total airborne radioactivity for all years. TRA releases dominated until the start of the RaLa process at ICPP, which was carried out primarily between 1957 and 1963. The effluent composition of the ANL-W reactor and processing facilities is similar to that at TRA and ICPP, but ANL-W activities resulted in the release of much smaller quantities. Releases from the LOFT facility, the only reactor operation at the TAN complex, consist of gaseous and particulate radionuclides generated during routine reactor operation but again in much smaller quantities than at TRA and ICPP.

Reprocessing operations did not begin at the ICPP until 1953, so discharges from the TRA comprised the majority of airborne releases in 1952. While release amounts were generally greater for the TRA from 1953 through 1956, the radionuclides released at the ICPP (^{131}I, ^{137}Cs, ^{90}Sr and ^{144}Ce) accounted for the greatest contribution to the screening dose. These results occurred because TRA releases included large amounts of ^{41}Ar and short-lived noble fission gases (e.g., xenon and krypton isotopes) that were important for the plume immersion pathway only. Production of the fission gas radionuclides was minimal at the ICPP because the short-lived gasses decayed appreciably in cooled fuel, and ^{41}Ar was produced through neutron activation of stable argon, a process occurring primarily in the reactors. The screening factors, which are based on the effective dose equivalent, for gaseous radioisotopes like ^{41}Ar were significantly lower than for particulate activity, such as ^{137}Cs and ^{90}Sr, releases which were generally higher at ICPP.

The RaLa process at ICPP from late 1956 through 1963 resulted in significant ^{131}I atmospheric releases. As iodine releases were reduced in 1958 and subsequent years following installation of charcoal beds, the relative dose from ICPP discharges also declined. Measured particulate emissions from the ICPP were reduced in 1975 following installation of the APS, which consisted of a fiberglass prefilter in series with HEPA filters.

For the episodic events and accidents, we evaluated and calculated relative screening values for a total of 134 individual episodic release events. The simplified screening methodology we used for the episodic events provided a comprehensive approach to evaluating each episodic release event for which it was possible to construct a source term. Because of the long duration and discontinuous nature of many of the release events, we divided the releases into four separate exposure categories, including single-day release, onsite exposure; single-day release, offsite exposure; multi-day event, onsite exposure; and multi-day event, offsite exposure. The release events that have high relative screening values in both onsite and offsite categories are the

- FEBT -B in 1957
- January 1961 SL-1 accident
- Ruthenium releases from the Waste Calcining Facility (WCF) at the ICPP in 1964
- IET-10 from late December 1957 through March 1958

- IET-4 from April to June 1956
- FECF filter break on October 30, 1956 at the ICPP
- October 1959 ICPP criticality
- Several RaLa releases: May 28, 1958; March 1, 1958; October 7, 1957

Because airborne releases from RaLa runs occurred for days to weeks after an operation, we included them as part of the routine releases. However, several RaLa runs that released significant amounts of material in a short time were also evaluated as episodic events.

A special exposure scenario concerning duck hunters was evaluated for potential exposure pathways of ingestion, exposure to external irradiation from ^{137}Cs contamination, and inhalation of airborne ^{137}Cs and $^{239,240}Pu$ contamination associated with plucking and using feathers in constructing a pillow. Ingestion dose calculations are reported here for ^{137}Cs, ^{134}Cs, ^{75}Se, ^{131}I, and $^{239,240}Pu$ contamination of muscle and liver, using average and maximum concentrations reported previously. The first three radionuclides had the highest concentrations measured in muscle tissue. Ingesting the duck meat was the most important exposure pathway and ^{137}Cs was the largest contributor to the ingestion dose. Based on these very conservative assumptions, we calculated the hunter's dose from eating one contaminated duck with the average measured ^{137}Cs concentration as 12 mrem (0.00012 Sv) (or 64 mrem [0.00064 Sv] if the maximum ^{137}Cs concentration is used in the calculations). This may be an important exposure pathway for some in the INEEL region.

Analysis of groundwater by the USGS over the years has resulted in the detection of a number of radioactive contaminants, including tritium, ^{90}Sr, ^{60}Co, ^{137}Cs, ^{129}I, ^{238}Pu, $^{239,240}Pu$, and ^{241}Am. We evaluated each of these radionuclides for movement offsite and as a potential exposure pathway. Except for tritium, the groundwater pathway was not considered to be a complete offsite exposure pathway for the other radionuclides for this historical screening analysis. Tritium concentrations in the groundwater were detected at the site boundary of the INEEL at different times during 1983-1985. We developed a scenario to assess the dose and risk associated with potential exposure to tritium in offsite groundwater. The dose for 1 year is equivalent 0.06 mrem (0.0000006 Sv). For perspective on the magnitude of this dose, we can compare this to the annual dose limit for drinking water exposure, which is 10 mrem. This analysis only indicates dose for offsite exposures to groundwater in the past and does not make any judgments regarding onsite exposures or future offsite exposures to other nuclides.

Our screening analysis work has identified some potential areas of consideration if additional resources and time were to be focused at the INEEL. Although it is difficult to rank these in a clear order and make direct comparisons because of different assumptions, release characteristics, and timing, we have attempted to develop a practical methodology for assessing the relative importance of all releases and evaluated exposure scenarios. To enable a defensible direct comparison between routine and episodic releases, we modified the episodic relative screening values and calculated screening values using the same process done for the routine releases. To accomplish this, we modified the relative screening values for the highest ranking episodic releases to evaluate each release in becquerels instead of curies, and we assumed a release duration period to provide an evaluation of average air concentrations instead of the total integrated concentration. We then modified this value by the ratio of the number of seconds in the assumed release duration period to the number of seconds in a year to account for the difference in release duration between the routine and episodic releases. This provided us with a set of

screening values that could be defensibly compared, both for onsite and offsite exposure scenarios (Tables 28 and 29). While this comparison enables a comprehensive evaluation of all releases and attempts to minimize biases created by using the NCRP air screening factors, it is still limited by the underlying assumptions of assumed annual average air concentrations and a 30-year buildup period

Combined, the episodic events we highlighted in that section are representative of the wide array of episodic-type releases that have occurred historically at the INEEL. More detailed dose evaluations for some number of the top releases in each category should enable a decision regarding the need for additional analyses for other, lower ranking releases. Several releases have high relative screening values in both onsite and offsite categories (e.g., FEBT-B, SL-1, WCF ruthenium releases, IET-10, IET-4, FECF filter break, 1959 ICPP criticality, and several RaLa releases), so a detailed analysis of one of these events would allow for an assessment of its overall importance in more than one category.

Routine releases from the ICPP, especially for the late 1950s, were also important for the INEEL region. A detailed evaluation of ICPP releases for one or more years (e.g., 1957, 1958, 1959) would allow for an assessment of health impacts related to both routine releases throughout the year as well as health impacts associated with shorter term releases, such as those related to individual RaLa runs or daily releases. In combination with detailed evaluations of a few other identified important episodic releases, particularly during these years, that address additional issues, such as the potential importance of short-lived radionuclides, release fraction uncertainties, and onsite exposures, a comprehensive determination about the relative importance of all INEEL releases could potentially be made.

The CDC, INEEL HES, and other involved stakeholders should continue to work closely to determine which release events, facilities, time periods, or radionuclides deserve further investigation into health impacts to potentially exposed members of the public, both at onsite and offsite locations.

Table 28. Ranking of episodic events and annual routine releases at offsite location[a]

Episode or year of routine releases[b]	Duration of release	Screening value		Percent of total screening value[c]
		Sv	mrem	
IET-10 – MD 1957	76 days	0.47	47000	59
IET-4 – MD 1956	73 days	0.045	4500	5.6
1957	1 year	0.041	4100	5.2
SL-1 accident – MD 1961	40 days	0.036	3600	4.6
1958	1 year	0.031	3100	3.9
FEBT-B-SD 1957	4 min	0.021	2100	2.6
1959	1 year	0.019	1900	2.4
RaLa Run #2 – MD 1957	34 days	0.0092	920	1.2
1965	1 year	0.0089	890	1.1
1956	1 year	0.0088	880	1.1
RaLa daily release – SD 1958	1 day	0.0083	830	1.1
RaLa daily release –SD 1958	1 day	0.0081	810	1.0
1955	1 year	0.0080	800	1.0
1963	1 year	0.0073	730	0.9
RaLa daily release –SD 1957	1 day	0.0068	680	0.9
1964	1 year	0.0058	580	0.7
1954	1 year	0.0058	580	0.7
1971	1 year	0.0057	570	0.7
Ruthenium releases SD 1964	1 day	0.0050	500	0.6
FECF Filter Break-SD 1958	1 hour	0.0047	470	0.6
1966	1 year	0.0044	440	0.6
1968	1 year	0.0038	380	0.5
1953	1 year	0.0032	320	0.4
1962	1 year	0.0025	250	0.3
1961	1 year	0.0021	210	0.3
Duck-eating 12 (average)[d]		0.0019	190	0.2
1969	1 year	0.0018	180	0.2
1974	1 year	0.0018	180	0.2
1960	1 year	0.0017	170	0.2
Duck-eating 1 (maximum)[d]		0.00094	94	0.1
Tritium in groundwater		0.0000006	0.06	0.0001

[a] Offsite location is Atomic City, 20 km from the ICPP and TRA. Person is located here all year for the routine releases, and for the duration of the episodic event and would be exposed through all pathways.

[b] For episodic events, the year of the episode or accident is given; MD indicates it occurred over multiple days and SD indicates a single day that year. For routine releases, the year is given.

[c] These events and routine release years contributed more than 99% of the total screening value at onsite locations; those that are highlighted represent more than 95% of the total screening value.

[d] (Average) indicates that we used the average measured concentrations of radionuclides in the calculations; (maximum) indicates we used the maximum concentration of radionuclides measured in waterfowl from the TRA ponds.

Table 29. Ranking of episodic events and annual releases at on onsite location[a]

Event or year[b]	Duration of episode	Screening value		Percent of total screening value[c]
		Sv	mrem	
SL-1 accident - MD 1961	40 days	0.015	1500	20
IET-10 -MD 1958	76 days	0.013	1300	17
1964	1 year	0.0039	390	5
1963	1 year	0.0031	310	4
1965	1 year	0.0028	280	4
1959 ICPP criticality -SD	15 min	0.0028	280	4
1955	1 year	0.0027	270	4
1966	1 year	0.0022	220	3
1956	1 year	0.0022	220	3
1953	1 year	0.0020	200	3
1954	1 year	0.0020	200	3
1961	1 year	0.0018	180	2
1962	1 year	0.0018	180	2
1958	1 year	0.0017	170	2
1959	1 year	0.0017	170	2
1957	1 year	0.0015	150	2
ICPP Pu release - MD 1959	2 days	0.0015	150	2
Ruthenium releases-SD 1964	1 day	0.0014	140	2
MTR stack release -SD 1963	1 day	0.0014	140	2
1960	1 year	0.0013	130	2
IET-4 MD 1956	73 days	0.0011	110	2
1967	1 year	0.0010	100	1
1968	1 year	0.0010	100	1
1952	1 year	0.00079	79	1
FECF Filter Break - SD 1958	1 hour	0.00075	75	1
Ruthenium releases -MD 1964	31 days	0.00042	42	0.6
1969	1 year	0.00041	41	0.6
1971	1 year	0.00039	39	0.5
FEBT-B - SD 1957	4 min	0.00026	26	0.3
1970	1 year	0.00025	25	0.3
RaLa Run #2 -MD 1957	34 days	0.00018	18	0.2
RaLa daily release - SD 1958	1 day	0.00015	15	0.2

[a] Onsite location at Highway 20, 6 km from the ICPP and the TRA. We assumed the person is located here all year for routine releases, and for the duration of the episodic release event and would be exposed through the inhalation and plume immersion pathways only.

[b] For episodic events, the year of the episode or accident is given; MD indicates it occurred over multiple days and SD indicates a single day that year. For routine releases, the year is given.

[d] These events and routine release years contributed more than 99% of the total screening value at onsite locations; those that are highlighted represent more than 95% of the total screening value.

REFERENCES

Anderson, E.L., and M.E. Weech. 1954. Status of Development of RaLa Process as of April 1, 1954. IDO-14300. INEEL Task Order Database MC Number 50184.

Anderson, E.L., R.S. MacCormack, and C.M. Slansky. 1959 Laboratory Development of a Process for Separating Barium-140 From MTR Fuel. IDO-14445. March 27. INEEL Task Order Database MC Number 50198.

Anderson, F.H. 1978. Unplanned Releases for 1977. FHA-86-78. April 7. INEEL Task Order Database MC Number 52220.

Anonymous. Date missing. Summary of Most Major ICPP Occurrences Which Are of Environmental Concern. INEEL Task Order Database MC Number 72386.

Anonymous. 1964. Waste Calciner Discharges. INEEL Task Order Database MC Number 60001.

Anonymous. 1981. Kr-85 Release. UOR#81-06. INEEL Task Order Database MC Number 50710.

ANL-W (Argonne National Laboratory-West). 1973. Facility Waste Descriptions Argonne-West. December. INEEL Task Order Database MC Number 50965.

Apostoaei, J., B.D. Blaylock, B. Caldwell, S. Flack, J.S. Gouge, F.O. Hoffman, C.J. Lewis, S.K. Nair, E.W. Reed, K.M. Thiessen, B.A. Thomas, and T.E. Widner. 1999. *Radionuclides Released to the Clinch River from White Oak Creek on the Oak Ridge Reservation–an Assessment of Historical Quantities Released, Off-Site Radiation Doses, and Health Risks.* Reports of the Oak Ridge Dose Reconstruction, Vol. 4. The Report of Project Task 4. July.

Ayers, A.L. and F. Burns. 1960. Waste Management at the Idaho Chemical Processing Plant. Paper for presentation at the American Nuclear Society Meeting, December 12-14, 1960. INEEL Task Order Database MC Number 73461.

Baker, R.E., C.C. Gamertsfelder, R.F. Gentzler. 1959. Final Report First Meltdown Experiment, Operation Boot. APEX-445. INEEL Task Order Database MC Number 51355.

Baker, R.E. 1961. Second Limited Melt Experiment (LIME-II): Final Report. General Electric. APEX-714. June 30. INEEL Task Order Database MC Number 51354.

Barraclough, J.T., B.D. Lewis, and R.G. Jensen. 1982. *Hydrologic Conditions at the Idaho National Engineering Laboratory, Idaho-Emphasis: 1974-1978.* U.S. Geological Survey, Water-Supply Paper 2191. INEEL Task Order Database MC Number 95358.

Bartholomay, R.C., B.R. Orr, M.J. Liszewski, and R.G. Jensen. 1995. *Hydrologic Conditions and Distributions of Selected Radiochemical and Chemical Constituents in Waste, Snake River Plain Aquifer, Idaho National Engineering Laboratory, Idaho 1989-1991.* U.S. Geological

Survey, Water-Resources Investigations Report 95-4175, DOE/ID-22123. INEEL Task Order Database MC Number 50933.

Berta, V.T. 1967. SNAPTRAN Experimental Measurements and Instrumentation. IDO-17205. INEEL Task Order Database MC Number 51363.

Black, D.E. and H.V. Chamberlain. Highlights of Meeting on Ruthenium Contamination. Bla-15-72 and Chm-1-72. February 11. INEEL Task Order Database MC Number 60255.

Boone, F.W., Lofthouse, J.H., and L.D. VanVleck. 1959. Gross Activity Stack Monitoring Results and Environmental Hazards Evaluation for the D101-L2A and D101L2C Test Series. DC-59-7-722. July 30. INEEL Task Order Database MC Number 51349.

Bowman, A.L., W.F. Downs, K.S. Moore, and B.F. Russell. 1984. INEL Environmental Characterization Report Volumes I and II. EGG-NPR-6688. September. INEEL Task Order Database MC Number 50394.

Bradley, T.M. 1991. Response to DOE-ID Request for NRF Effluent Data for INEL Historical Dose Evaluation. NR:IBO-91/117. April 10. INEEL Task Order Database MC Number 73020.

Brodsky, A. and G.V. Beard (eds.). 1960. A Compendium of Information for Use in Controlling Radiation Emergencies, Including Lecture Notes from a Training Session in Idaho Falls, ID, February 12-14, 1958. ID-8206. September. INEEL Task Order Database MC Number 52221.

Brown, R.A., J.D. Christian, and T.R. Thomas. 1983. Airborne Radionuclide Waste Management. ENICO-1132. June. INEEL Task Order Database MC Number 73481.

Buckham, J.A. 1972. Status of Ruthenium Contamination Studies. Buc-44-72. February 14. INEEL Task Order Database MC Number 60267.

Bunch, D.F. 1965. SPERT I Destructive Test Series Environmental Monitoring and Research Studies. IDO-12039. INEEL Task Order Database MC Number 51362.

Bunch, D.F. (ed.). 1966. Controlled Environmental Radioiodine Tests, Progress Report Number Two. IDO-12053. August. INEEL Task Order Database MC Number 51361.

Bunch, D.F. (ed.). 1968. Controlled Environmental Radioiodine Tests, Progress Report Number Three. IDO-12063. January. INEEL Task Order Database MC Number 51360.

Burgus, W.H. 1959. Radiochemical Analyses of CPP Materials. Bur-43-59A. October 28. INEEL Task Order Database MC Number 50181.

Carboneau, M.L. 1987. OECD LOFT Fission Product Experiment LP-FP-2 Fission Product Data Report. OECD LOFT-T-3805. May. INEEL Task Order Database MC Number 51769.

Casto, W.R. 1980. ICPP Criticality Event of October 17, 1978. *Nuclear Safety* 21 (5): 648-653.

Cederberg, G.K., and J.R. Bower. 1959. Minimizing Gaseous and Particulate Activity in Idaho chemical Processing Plant Off-gas. PTR-418. July 2. INEEL Task Order Database MC Number 73458.

Cederberg, G.K., and D.K. MacQueen. 1961. Containment of Iodine-131 Released by the RaLa Process. IDO-14566. INEEL Task Order Database MC Number 50199.

Cember, H. 1988. *Health Physics*. Second edition. Pergamon Press.

Chew, E.W. and R.G. Mitchell. 1988. 1987 Environmental Monitoring Program Report for the Idaho National Engineering Laboratory Site. DOE/ID-12082(87). May. INEEL Task Order Database MC Number 50966.

Clawson, K.L., G.E. Start, and N.R. Ricks. 1989. Climatography of the Idaho National Engineering Laboratory, 2nd Edition. DOE/ID-12118. December. INEEL Task Order Database MC Number 73021.

Commander, R.E. 1974. Release of Activity from CPP Stack in Excess of Shutdown Reporting Limit. SOOR-74-15. July 3. INEEL Task Order Database MC Number 52141.

Convair (Division of General Dynamics Corporation). 1959. Fission Products Field Release Test-I. U.S. Air Force Nuclear Aircraft Research Facility. NARF-59-32T. September. INEEL Task Order Database MC Number 52151.

Cordes, O.L., J.R. Fielding, G.A. Dinneen, and R.P. Bird. 1965. Radiological Aspects of SNAPTRAN 2/10-3 Destructive Test. IDO-17038. INEEL Task Order Database MC Number 51359.

Cordes, O.L., D.F. Bunch, J.R. Fielding, and J.K. Warkentin. 1967. Radiological Aspects of the SNAPTRAN-2 Destructive Test. IDO-17203. February. INEEL Task Order Database MC Number 51358.

De Boisblanc, D.R. 1958. Testimony of DesLonde DeBlanc regarding the Materials Testing Reactor at the TRA. Task Order Database MC Number 73456.

De Boisblanc, D.R. 1969. Reactor Engineering Branch Annual Report, Fiscal Year 1968. IN-1228. February. INEEL Task Order Database MC Number 73050

Devens, F.G., G.D. Pincock, G. St. Leger-Barter, and L.G. Whitlow. 1958. Operation BOOT Test Results (IET #12). DC 58-7-728. July 27. INEEL Task Order Database MC Number 73041.

Devens, F.G. No date. Data Report-D102 A2 Power Plant Testing-IET 13. DC 59-4-410. INEEL Task Order Database MC Number 73040

Dickey, B. R., W. F. Holcomb, L. T. Lakey, R. D. Modrow, C. M. Slansky. 1972. Plan for the Management of Radioactive Wastes at the Idaho Chemical Processing Plant. ACI-123. July. INEEL Task Order Database MC Number 51027.

Dickson, C.R. and P.G. Voillequé. 1972. NRTS Long Distance Diffusion Tests. September 8. INEEL Task Order Database MC Number 73022.

Dietrich, J.R. 1954. Experimental Investigation of the Self-Limitation of Power During Reactivity Transients in a Subcooled, Water-Moderated Reactor, Borax-I Experiments, 1954. U.S. Atomic Energy Commission. AECD-3668. June 19. INEEL Task Order Database MC Number 50203.

DOE (Department of Energy). 1979. 1978 Environmental Monitoring Program Report for the Idaho National Engineering Laboratory Site. IDO-12082(78). May. INEEL Task Order Database MC Number 50925.

DOE. 1980. 1979 Environmental Monitoring Program Report for the Idaho National Engineering Laboratory Site. IDO-12082(79). May. INEEL Task Order Database MC Number 50926.

DOE. 1981. 1980 Environmental Monitoring Program Report for the Idaho National Engineering Laboratory Site. IDO-12082(80). May. INEEL Task Order Database MC Number 50927.

DOE. 1982. 1981 Environmental Monitoring Program Report for the Idaho National Engineering Laboratory Site. IDO-12082(81). May. INEEL Task Order Database MC Number 50238.

DOE. 1983. 1982 Environmental Monitoring Program Report for the Idaho National Engineering Laboratory Site. IDO-12082(82). May. INEEL Task Order Database MC Number 50929.

DOE. 1991a. Idaho National Engineering Laboratory Historical Dose Evaluation: Volume 1 and Appendices: Volume 2. DOE/ID-12119. August. INEEL Task Order Database MC Number 51061.

DOE. 1991b. Idaho National Engineering Laboratory Historical Dose Evaluation Quality Assurance Files. DOE/ID-12119-QAF-001-128. INEEL Task Order Database MC Number 73023.

Dykes, J.W., J.D. Ford, K.R. Hoopingarner. 1965. A Summary of the 1962 Fuel Element Fission Break in the MTR. IDO-17064. February. INEEL Task Order Database MC Number 73045.

Eichholz, G.G. 1983. "Source Terms for Nuclear Facilities, and Medical and Industrial Sties." Radiological Assessment, A Textbook on Environmental Dose Analysis. Edited by J.E. Till and H. R. Meyer. Pages 1-1–1-54. September.

ENICO (Exxon Nuclear Idaho Company, Inc.) 1983a. ICPP 1st-Quarter 1983 Effluent Monitoring Report. ENICO-1116 Supplement 1. August. INEEL Task Order Database MC Number 51096.

ENICO (Exxon Nuclear Idaho Company, Inc.) 1983b. ICPP 2nd Quarter 1983 Effluent Monitoring Report. ENICO-1116 Supplement 1. October. INEEL Task Order Database MC Number 51097.

EPA (U.S. Environmental Protection Agency). 1991. *Role of the Baseline Risk Assessment in Superfund Remedy Selection Decisions.* Memorandum from Assistant Administrator Don R. Clay to the Regions. OSWER Directive 9355.0-30. Office of Solid Waste and Emergency Response, EPA, Washington, DC. April 22.

EPA. 1999. *Cancer Risk Coefficients for Environmental Exposure to Radionuclides.* U.S. EPA, Office of Radiation and Indoor Air, Federal Guidance Report No. 13, EPA 402-R-99-001. September.

ERDA (Energy Research and Development Administration). 1977. Final Environmental Impact Statement, Waste Management Operations. Idaho National Engineering Laboratory. ERDA-1536. September. INEEL Task Order Database MC Number 73051.

Evans, R.C. 1957. HTRE-2 Insert-1-Operations Report. XDC 57-10-26. October 1. INEEL Task Order Database MC Number 51364.

Evans, R.C. 1958. HTRE-2 Insert 1-C (D101-C3) Operations Report. XDC 58-9-199. September 22. INEEL Task Order Database MC Number 73042.

Evans, R.C. 1959. Operations Report, HTRE-2, L2C-1 Cartridge. XDC 59-11-179. November 16. INEEL Task Order Database MC Number 73043.

Evans, R.C. 1960. HTRE-2 Insert L2E-1 Cartridge Operations Report. XDC 60-3-111. March 8. INEEL Task Order Database MC Number 51366.

Field, J.H. (ed.), D.C. Foster, J.H. Field, J.R. Elms, and F.L. Sims. 1961. Test Results from D101-L2E-6 Power Testing, IET #26. DC-61-6-704. INEEL Task Order Database MC Number 51367.

Flagella, P.N. 1962 Heat Transfer Reactor Experiment No. 2. APEX-905. May 25. INEEL Task Order Database MC Number 51368.

Foster, D.C., J.W. Highberg, S.W. Gabriel, N.K. Sowards. 1958. Power Testing Results from D 101 D2 Core (IET #10). XDCL 58-7-715. July 3. INEEL Task Order Database MC Number 51370.

Foster, D.C., C.N. Anderson, H.E. Anderson, C.A. Gukeisen, and G.D. Pincock. 1960. Results from the Power Testing of D101-L2E-2, IET 20. DC 60-9-722. September 19. INEEL Task Order Database MC Number 51371.

Francis, W.C. and F.H. Tingey. 1967. Quarterly Report, Status of Test Reactor Fuel Element Activities, Period Ending April 30, 1967. Idaho Nuclear Corporation. CI-1052. INEEL Task Order Database MC Number 73049.

Fromm, J. 1996. INEEL Environmental Toxicologist, Remediation Bureau. Memo to INEEL WAG Managers and Technical Support Staff. Subject: Radionuclide Risk-Based Concentration Tables. January 3. INEEL Task Order Database MC Number 73480.

Gammill, W.P. 1961. Environmental Aspects of the SL-1 Accident. 7th Annual Meeting of the Pacific Northwest Section of the American Industrial Hygiene Association, September 22, 1961. INEEL Task Order Database MC Number 50205.

GE (General Electric). 1959. Summary Report of HTRE No. 3 Nuclear Excursion. Aircraft Nuclear Propulsion Department. APEX-509. August. INEEL Task Order Database MC Number 51372.

GE. 1962. Final Report of SL-1 Recovery Operation, May 1961 Through July 1962. Flight Propulsion Laboratory Department, IDO-19311. July 27. INEEL Task Order Database MC Number 51353.

Gibson, G.W., M.J. Graber, W.C. Francis. 1963. Annual Progress Report on Fuel Element Development for FY 1963. IDO-16934. November 25. INEEL Task Order Database MC Number 73048.

Ginkel, W.L., F.H. Tingey, C.W. Bills, A.O. Dodd, and K.K. Kennedy. 1960. Nuclear Incident at the Idaho Chemical Processing Plant. IDO-10035. February 15. INEEL Task Order Database MC Number 50181.

Halford, D. K., O. D. Markham, and R.L. Dickson. 1981. Radiation Doses to Waterfowl Using a Liquid Radioactive Waste Disposal Area. *J. Wildlife Management* 46(4): 905-914.

Halford, D. K., D. Markham, and G.C. White. 1983. Biological Elimination Rates of Radioisotopes by Mallards Contaminated at a Liquid Radioactive Waste Disposal Area. *Health Physics* 45 (3): 745-756.

Hawley, C.A., Jr. (ed.). 1964. Controlled Environmental Radioiodine Tests at the National Reactor Testing Station. IDO-12035. June. INEEL Task Order Database MC Number 51365.

Hayden, R.E. 1957-1963. Activity Discharged to Atmosphere. INEEL Task Order Database MC Number 60111.

Hayden, R.E. 1958. Incident Report. Ha-46-58A. March 26. INEEL Task Order Database MC Number 60253.

Hayden, R.E. and B.L. Rich. 1958-1959. Series of notes and memos related to the ICPP/FECF particulate release problem. Task Order Database MC Number 73468.

Hayden, R.E. 1959. ICPP Incident – October 16, 1959: Summary of Data from Fifty-four Recording Radiation Detection Instruments by the CPP Health Physics Section. Ha-180-59A. November 3. INEEL Task Order Database MC Number 50185.

Hayes, D.F. 1956. A Summary of Accidents and Incidents Involving Radiation in Atomic Energy Activities, June 1945 through December 1955. TID-5360. August. INEEL Task Order Database MC Number 51373.

Hayes, D.F. 1959. A Summary of Industrial Accidents in USAEC Facilities. TID-5360 (Suppl. 2). September. INEEL Task Order Database MC Number 73036.

Highberg, J.W., R.F. Brenton, C.B. Cannon, K.J. Skow, D.A. McKenzie, C. Davidson, J.F. Kunze, and A.F. Clark. 1960. Preliminary Data Report IET Test Series No. 18 Phase II Testing of the D102A2 Power Plant. DC-60-6-735. June 17. INEEL Task Order Database MC Number 51374.

Highberg, J.W., C.B. Cannon, J.H. Field, K.J. Skow, and C. Davidson. 1961. Preliminary Data Report IET Test Series No. 25 D102A2 Power Plant. DC-61-2-724. February 10. INEEL Task Order Database MC Number 51375.

Hoff, D.L., E.W. Chew, and R.L. Dickson. 1984. 1983 Environmental Monitoring Program Report for the Idaho National Engineering Laboratory Site. DOE/ID-12082(83). May. INEEL Task Order Database MC Number 50930.

Hoff, D.L., E.W. Chew, and R.L. Dickson. 1985. 1984 Environmental Monitoring Program Report for the Idaho National Engineering Laboratory Site. DOE/ID-12082(84). May. INEEL Task Order Database MC Number 50949.

Hoff, D.L., E.W. Chew, and S.K. Rope. 1986. 1985 Environmental Monitoring Program Report for the Idaho National Engineering Laboratory Site. DOE/ID-12082(85). May. INEEL Task Order Database MC Number 50946.

Hoff, D.L., E.W. Chew, and S.K. Rope. 1987. 1986 Environmental Monitoring Program Report for the Idaho National Engineering Laboratory Site. DOE/ID-12082(86). May. INEEL Task Order Database MC Number 50956.

Hoff, D.L., R.G. Mitchell, and R. Moore. 1989. The Idaho National Engineering Laboratory Site Environmental Report for Calendar Year 1988. DOE/ID-12082(88). June. INEEL Task Order Database MC Number 50938.

Hoff, D.L., R.G. Mitchell, G.C. Bowman, and R. Moore. 1990. The Idaho National Engineering Laboratory Site Environmental Report for Calendar Year 1989. DOE/ID-12082(89). June. INEEL Task Order Database MC Number 50936.

Hoff, D.L., R.G. Mitchell, R. Moore, and R.M. Shaw. 1991. The Idaho National Engineering Laboratory Site Environmental Report for Calendar Year 1990. DOE/ID-12082(90). June. INEEL Task Order Database MC Number 50937.

Hoff, D.L., R.G. Mitchell, R. Moore, and L. Bingham. 1992. The Idaho National Engineering Laboratory Site Environmental Report for Calendar Year 1991. DOE/ID-12082(91). September. INEEL Task Order Database MC Number 50951.

Hoff, D.L., R.G. Mitchell, R. Moore, and L. Bingham. 1993. The Idaho National Engineering Laboratory Site Environmental Report for Calendar Year 1992. DOE/ID-12082(92). June. INEEL Task Order Database MC Number 50914.

Hogg, G.W., W.F. Holcomb, L.T. Lakey, L.H. Jones, and D.D. Coward. 1971a. A Survey of NRTS Waste Management Practices. ICP-1042-Volume 1. September. INEEL Task Order Database MC Number 50395.

Hogg, G.W., W.F. Holcomb, L.T. Lakey, L.H. Jones, and D.D. Coward. 1971b. A Survey of NRTS Waste Management Practices. ICP-1042-Volume II. September. INEEL Task Order Database MC Number 50395.

Holcomb, W.F. and A.P. Larrick. 1974. Effective Water Utilization at the NRTS Test Reactor Area. ACI-159. February. INEEL Task Order Database MC Number 50957.

Honkus, R. J. M. 1982. ICPP CY-1981 Effluent Monitoring Report. ENICO-1115. May. INEEL Task Order Database MC Number 73473.

Horan, J.R. 1962. Annual Report of Health and Safety Division, 1961. Idaho Operations Office. IDO-12021. May 21. INEEL Task Order Database MC Number 52157.

Horan, J.R. 1963. Annual Progress Report – 1962, Health and Safety Division. Idaho Operations Office. IDO-12033. May 21. INEEL Task Order Database MC Number 52158.

Horan, J.R. 1966. Iodine Release at CPP – October 31, 1966. November 7. INEEL Task Order Database MC Number 50773.

Hull, L.C. 1989. Conceptual Model and Description of the Affected Environment for the TRA Warm Waste Pond (Waste Management Unit TRA-03). Informal Report. EGG-ER-8644. October. INEEL Task Order Database MC Number 51054.

ICRP. 1995. *Age-Dependent Doses to Members of the Public from Intake of Radionuclides: Part 5 Compilation of Ingestion and Inhalation Dose Coefficients.* ICRP Publication 72. International Commission of Radiological Protection. September.

Ikenberry, T.A. 1984a. Estimate of Radioactive Release from 5th CSSF Bin. TAI-06-84. July 20. INEEL Task Order Database MC Number 50558.

Ikenberry, T.A. 1984b. Revised Estimate of Release from 5th CSSF Bin. TAI-08-84. August 7. INEEL Task Order Database MC Number 50559.

INEL (Idaho National Engineering Laboratory). 1979. 1978 Environmental Monitoring Program Report for Idaho National Engineering Laboratory Site. IDO-12082(78). May. INEEL Task Order Database MC Number 50925.

INEL. 1983. 1982 Environmental Monitoring Program Report for Idaho National Engineering Laboratory Site. DOE/ID-12082(82). May. INEEL Task Order Database MC Number 50929.

Islitzer, N.F. 1962. The Role of Meteorology Following the Nuclear Accident in Southeast Idaho. IDO-19310. May. INEEL Task Order Database MC Number 51356.

Johnson, B.D. 1960. Report of Radiation Incident – MTR Stack. May 17. INEEL Task Order Database MC Number 71209.

Johnson, B.D. 1963. MTR Health Physics Report for February, 1963. Joh-7-63A. INEEL Task Order Database MC Number 55197 and 75014.

Keller, F.R. 1962. Fuel Element Flow Blockage in the Engineering Test Reactor. IDO-16780. May 10. INEEL Task Order Database MC Number 73044.

Keller, J.H. 1974a. Stack Releases. JHK-1-74. October 8. INEEL Task Order Database MC Number 52133.

Keller, J.H. 1974b. October Stack Releases. JHK-8-74. November 7. INEEL Task Order Database MC Number 52134.

Kerr, W.B. 1971. A Study of Radioactive Liquid Waste Management at Test Area North (TAN). ACI-101. December. INEEL Task Order Database MC Number 73479.

King, W.C. 1956. Monitoring the Liquid Wastes Discharged from the Idaho Chemical Processing Plant. IDO 14378. Phillips Petroleum Company. June 12. INEEL Task Order Database MC Number 50201.

Krivanek, K.R. 1988. ICPP Environmental Monitoring Report CY-1986. WINCO-1052. February. INEEL Task Order Database MC Number 51085.

Kunze, J.F. 1962. Additional Analysis of the SL-1 Excursion, Final Report of Progress July through October 1962. IDO-19313. November 21. INEEL Task Order Database MC Number 51357.

Lakey, L. T., J. R. Bower, J. A. McBride, J. A. Buckham, J. R. Huffman. 1963. ICPP Waste Calcining Facility Safety Analysis Report. IDO-14620-Rev-1. December. INEEL Task Order Database MC Number 50960.

Legler, B.M., W.L. Chase, M.D. Martin, D.M. Paige, F.K. Wrigley. 1955. Pilot Plant Development of a Process for Separating Barium-140 from MTR Fuel. IDO-14344. September 1. INEEL Task Order Database MC Number 73478.

Legler, B.M., S.F. Fairbourne, P.N. Kelly, R.A. Robinson. 1957. Startup Operation of a Production Facility for Separating Barium-140 From MTR Fuel. IDO-14414. September 1. INEEL Task Order Database MC Number 50197.

Lewis, W.B. 1960. Analysis of Critical Incident at ICPP. WBL-6-60A-M. February 19. INEEL Task Order Database MC Number 50181.

Lewis, W.B. and R.G. Jansen 1984. Hydrologic Conditions at the Idaho National Engineering Laboratory, Idaho: 1971-1981 Update. U.S. Geological Survey Open-File Report 84-230. IDO-22066. INEEL Task Order Database MC Number 73477.

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

Litteer, D.L., C.N. Peterson, and A.M. Sims. 1991. Radioactive Waste Management Information for 1990 and Record-To-Date. DOE/ID-10054(90). Idaho Falls, Idaho: EG&G Idaho, Inc. July.

Litteer, D.L. and B.D. Reagan. 1989. Radioactive Waste Management Information for 1988 and Record-To-Date. DOE/ID-10054(88). Idaho Falls, Idaho: EG&G Idaho, Inc. July.

Markham, O.D., D. K. Halford, S.K. Rope, and G.B. Kuzo. 1988. Plutonium, Am, Cm, and Sr in Ducks Maintained on Radioactive Leaching Ponds in Southeastern Idaho. *Health Physics* 55(3): 524-517-524.

Marter, W.L. 1965. Nuclear Accidents in the United States 1945-1965. INEEL Task Order Database MC Number 71526.

McGavran, P.D. and M.C. Case. 1999. *The Feasibility of Performing a Chemical Dose Reconstruction Study at the INEEL.* RAC Report No. 4-CDC-Task Order 1-1999-Final. Risk Assessment Corporation, Neeses, South Carolina.

Mikkola, B.J. 1988. Followup Report on Idaho Chemical Processing Plant Radioactive Particulate Release. November 1. INEEL Task Order Database MC Number 50746 and 51306.

Miller, C.L., R.N. Nassano, W.C. Powell. 1960. Interim Report on HTRE No. 3 (D102A2) Operations. DC 60-1-50. January 7. INEEL Task Order Database MC Number 51377.

Miller, R.W., A. Sola, and R.K. McCardell. 1964. Report of the SPERT I Destructive Test Program on an Aluminum, Plate-Type, Water-Moderated Reactor. IDO-16883. June. INEEL Task Order Database MC Number 51378.

NCRP. (National Council on Radiation Protection and Measurements.) 1996. *Screening Models for Releases of Radionuclides to Atmosphere, Surface Water, and Ground.* NCRP Report No. 123. National Council on Radiation Protection and Measurements. Bethesda, MD. January.

Nebeker, R.L. and L.T. Lakey 1970. A Study of Possible Improvements in Liquid Waste Management at the NRTS Test Reactor Area. CI-1173. Idaho Nuclear Corporation National Reactor Testing Station. April. INEEL Task Order Database MC Number 73474.

O'Neill, L.J. 1966. Suspected I-131 Release from Expended Core Facility, 11/7/66. November 16. INEEL Task Order Database MC Number 75071.

Orr, B.R. and L.D. Cecil. 1991. *Hydrologic Conditions and Distribution of Selected Chemical Constituents in Water, Snake River Plain Aquifer, Idaho National Engineering Laboratory, Idaho, 1986 to 1988.* U.S. Geological Survey, Water-Resources Investigations Report 91-4047. INEEL Task Order Database MC Number 50932.

Osloond, J.H. 1970. Waste Disposal Data for the National Reactor Testing Station. IDO-12074. INEEL Task Order Database MC Number 73475.

Paulus, R.C, F.M. Warzel, F.H. Tingey, A.O. Dodd, and K.K. Kennedy. 1961. Nuclear Incident at the Idaho Chemical Processing Plant, January 25, 1961. IDO-10036. June. INEEL Task Order Database MC Number 51387.

Pincock, G.D. (ed.). 1959. Test Results from D 101 L2A-1 Power Testing, IET #14. DC 59-8-728. August 19. INEEL Task Order Database MC Number 51379.

Pincock, G.D. 1960a. Test Results from D101-L2E-1 Power Testing, IET #17. DC 60-4-700. March 31. INEEL Task Order Database MC Number 51381.

Pincock, G.D. 1960b. Test Results from D101-L2E-3 Power Testing, IET #19. DC 60-7-717. July 8. INEEL Task Order Database MC Number 51382.

Pincock, G.D. 1960c. Test Results from the D101-L2A-2 Fuel Element Effluent Test (FEET), IET 21 and 23. General Electric, Aircraft Nuclear Propulsion Department. DC 60-12-732. December 21. INEEL Task Order Database MC Number 51384.

Pincock, G.D. 1960d. Test Results from D101-L2E-4 Lime Experiment, IET 22. General Electric, Aircraft Nuclear Propulsion Department. DC 60-10-720. October 12. INEEL Task Order Database MC Number 51383.

Pincock, G.D. 1960e. Test Results from the D101-L2E-5 Sub-Lime Experiment IET 24. General Electric, Aircraft Nuclear Propulsion Department. DC 60-11-719. November 23. INEEL Task Order Database MC Number 51380.

Pittman, J.R., R.G. Jensen, and P.R. Fischer. 1988. *Hydrologic Conditions at the Idaho National Engineering Laboratory, 1982 to 1985.* U.S. Geological Survey, Water-Resources Investigations Report 89-4008, DOE/ID-22078. INEEL Task Order Database MC Number 50918.

Rich, B.L. 1959. Radioactivity Discharge from FECF Incident. BLR-6-59A. January 8. INEEL Task Order Database MC Number 71225.

Rich, B.L. 1960. Report of Radiation Incident – ETR Stack. July 1. INEEL Task Order Database MC Number 71204.

Rich, B.L. 1962. ETR Health Physics Monthly Report for December 1961. BLR-9-62A. January 31. Task Order Database MC Number 71220.

Rigstad, N.J. 1958. Cell U Iodine Release Incident. Rig-15-58A. March 24. INEEL Task Order Database MC Number 60253.

Ritter, G.L. 1981. Letter to S.C. Vorndran: Projected Composition of Waste Going to Injection Well. GLR-53-81. March 25. Task Order Database MC Number 51076.

Ruhter, P.E. 1970. Experimental Cloud Exposure Study (EXCES). March 6. INEEL Task Order Database MC Number 50576.

SC&A (Sanford Cohen and Associates, Inc.). 1994. Identification, Retrieval and Evaluation of documents and Data Pertinent to a Historical Dose Reconstruction and The INEL. Final Report. Prepared by Sanford Cohen and Associates, Shonka Research Associates, and Woodward-Clyde Consultants for the Centers for Disease Control and Prevention, December 21.

Shleien, B.J. 1992. Phase I Report: Inventory Values *Scoping Document for Determination of Temporal and Geographic Domains for the HEDR Study.* Hanford Technical Steering Panel.

Showalter, D.E. (ed.), C. Davidson, D.A. McKenzie, J.W. Highberg, K.J. Skow, R.F. Brenton, D.C. Foster, and G. St. Leger-Barter. 1959. Preliminary Data Report IET 16 D102A2(A). SDCL 59-11-715. October 28. INEEL Task Order Database MC Number 51386.

Smith, E.H. 1962. MTR Progress Report, Cycle No. 182, October 29 – November 19, 1962. IDO-16831. December 7. INEEL Task Order Database MC Number 73046.

Smith, C.J. 1981. A History of the Radioactive Waste Management Complex at the Idaho National Engineering Laboratory. WM-F1-81-003. Box 83(A6). March. INEEL Task Order Database MC Number 50827.

Sommers, J.F. 1958a. GEH-4 Fission Break Incident, May 1-2, 1958. Som-54-58A. May 7. INEEL Task Order Database MC Number 60253.

Sommers, J.F. 1958b. Incident Report on GEH-4 Rupture on 12-17-58. Som-126-58A. December 23. INEEL Task Order Database MC Number 60253.

Stachew, J.C. 1985. Iodine Release from LOFT as of July 11, 1985. July 11. INEEL Task Order Database MC Number 51329.

Start, G.E. 1966. 1430 MST ECF Release, 11-7-66. November 9. INEEL Task Order Database MC Number 75071.

Stevenson, C.E. 1957. Technical Progress Report for April Through June 1957 Idaho Chemical Processing Plant. IDO-14419. Phillips Petroleum Co. September 25. INEEL Task Order Database MC Number 73476.

Thornton, G., S.H. Minnich, C. Heddleson, and D.H. Culver. 1962. Heat Transfer Reactor Experiment No. 1. General Electric, Direct-Air-Cycle Aircraft Nuclear Propulsion Program. APEX-904. February 28. INEEL Task Order Database MC Number 51389.

Till, J.E. 1984. "Source Terms for Technetirum-99 from Nuclear Fuel Cycle Facilities," in *Technetium in the Environment*. Ed. By G. Desmet and C. Myttenacre. Elvesier Applied Science Publishers, New York.

Trojanowski, R.E. 1974. Radioactive Effluent Report 1953 through 1973 for the Naval Reactors Facility. NRF-RS-RCE-186. April 23. INEEL Task Order Database MC Number 50183.

USAEC (United States Atomic Energy Commission). 1959. Annual Report – 1958, Health and Safety Division. Idaho Operation Office. IDO-12012. INEEL Task Order Database MC Number 52154.

USAEC. 1960. Annual Report of Health and Safety Division, 1959. Idaho Operations Office. IDO-12014. October. INEEL Task Order Database MC Number 52155.

USAEC. 1961. Annual Report of Health and Safety Division, 1960. Idaho Operations Office. IDO-12019. September. INEEL Task Order Database MC Number 52156.

USAEC. 1963. A Summary of Industrial Accidents in USAEC Facilities 1961-1962. TID-5360 (Suppl. 4). December. INEEL Task Order Database MC Number 73034.

USAEC. 1965. A Summary of Industrial Accidents in USAEC Facilities 1963-1964. TID-5360 (Suppl. 5). December. INEEL Task Order Database MC Number 73035.

USAEC. 1967a. A Summary of Industrial Accidents in USAEC Facilities 1956. TID-5360 (Suppl. 1, Rev.). August. INEEL Task Order Database MC Number 73037.

USAEC. 1967b. A Summary of Industrial Accidents in USAEC Facilities 1965-1966. TID-5360 (Suppl. 6). December. INEEL Task Order Database MC Number 73038.

USAEC. 1971. Airborne and Liquid Radioactive Waste Management Information Summary for 1962 through 1970 at the Argonne National laboratory-West, Chemical Processing Plant, Naval Reactors Facility, Test Reactors Area Located at the National Reactor Testing Station, Idaho. IDO-100544(62-70). Idaho Operations Office. INEEL Task Order Database MC Number 50184.

USAEC. 1975. Operational Accidents and Radiation Exposure Experience within the United States Atomic Energy Commission 1943-1975. WASH-1192. INEEL Task Order Database MC Number 73039.

Voillequé, P.G. 1969. Outline of Plans for EXCES Xe-133 Releases Tests. August 13. INEEL Task Order Database MC Number 50575.

Voillequé. P.G. 1971. Addendum Number 1 to Brief Environmental Statement for NRTS Long Distance Diffusion Tests (Phase I). October 4. INEEL Task Order Database MC Number 50732.

Volpe, J.J., K.F. Childs, W.A. Mickelson, B.W. Garrow, and W.G. Robson. 1988. Investigation Report of the Ruthenium Release During NWCF Calcine Bed Dissolution, October 30, 1988 at INEL CPP-659. Westinghouse Idaho Nuclear Company. November 10. INEEL Task Order Database MC Number 51305.

Warzel, F.M. 1961. Investigation Committee Report – ETR Blockage by Sight Box on December 12, 1961. Wrz-71-61A. December 29. INEEL Task Order Database MC Number 71203.

Wehman, G. 1959. Preliminary Report of Fission Products Field Release Test-I. IDO-12006. February. INEEL Task Order Database MC Number 51347.

Wehman, G. 1972. Minutes of Eighth Meeting of the Ad Hoc Task Force on Ruthenium Particle Problem. May 23. INEEL Task Order Database MC Number 60267.

Wenzel, D.R. 1982. RSAC-3, Radiological Safety Analysis Computer Program. ENICO-1002. INEEL Task Order Database MC Number 73024.

Wenzel, D.R. 1990. Interim Users Manual for RSAC-4, Radiological Safety Analysis Computer Program, Version 4.03. Westinghouse Idaho Nuclear Company, Inc. April 10. INEEL Task Order Database MC Number 73024.

Wenzel, D.R. 1994. Radiological Safety Analysis Computer Program and User's Manual for RSAC-5. WINCO-1123. February. INEEL Task Order Database MC Number 73024.

Wheeler, B.R. 1959. Waste Calcination Off-Gas Studies. Phillips Petroleum Company. PTR-417. July 2. INEEL Task Order Database MC Number 73459.

Wilks, P.H. 1959. The Release of Fission Products from Reactor Fuel, a Literature Survey. APEX-474. March 23. INEEL Task Order Database MC Number 52153.

Williamson, M.M. 1977. Above-Normal Release of Radionuclides by ICPP, November 14, 1977. December 9. INEEL Task Order Database MC Number 50719.

WINCO (Westinghouse Idaho Nuclear Company, Inc.). 1984. ICPP Fourth Quarter CY-1983 Effluent Monitoring Report. WINCO-1002 Supplement 1. February. INEEL Task Order Database MC Number 51099.

WINCO (Westinghouse Idaho Nuclear Company, Inc.). 1985. ICPP Fourth Quarter 1984 Effluent Monitoring Report. WINCO-1002 Supplement 1. February. INEEL Task Order Database MC Number 51082.

WINCO (Westinghouse Idaho Nuclear Company, Inc.). 1986. ICPP Environmental Monitoring Report fourth Quarter CY-1985. WIN-181. March. INEEL Task Order Database MC Number 51084.

WINCO (Westinghouse Idaho Nuclear Company, Inc.). 1991. ICPP Environmental Monitoring Report CY-1989. WINCO-1085. January. INEEL Task Order Database MC Number 51089.

Wong, M.A. and R.S. Roberts. 1979. Description of the Stack Monitoring System and Effluent Measuring Techniques at the Idaho Chemical Processing Plant. ENICO-1005. August. INEEL Task Order Database MC Number 73472.

Zimbrick, J.D. and P.G. Voillequé (eds.). 1969. 1967 CERT Progress Report, Controlled Environmental Radioiodine Tests at the National Reactor Testing Station, Progress Report Number Four. IDO-12065. INEEL Task Order Database MC Number 51391.

APPENDIX A: RELEASE POINTS FROM INEEL FACILITIES

Table A1. Major Release Points to the Environment from INEEL Facilities[a]

Type of discharge	Area or location	Type and description of discharge structure
Airborne	Auxiliary Reactor Area	9.1 m stack; 1.4 $m^3 s^{-1}$ discharge capacity; continuously monitored
	Argonne National Lab-West	
	Experimental Breeder Reactor -II	61 m glass-coated steel stack; 31.2 $m^3 s^{-1}$ discharge capacity; continuously monitored. Cooling tower: circulation rate of 7.6×10^4 liters per month. Drift rate 0.01%; blowdown rate of 100-190 liters per month
	FASB	10 m stack; 7.8 $m^3 s^{-1}$ discharge capacity; continuously monitored
	Hot Fuel Examination Facility	28.6 m stack; 20.3 $m^3 s^{-1}$ discharge capacity; continuously monitored
	SCCF	14.6 m stack; 4.7 $m^3 s^{-1}$ discharge capacity; continuously monitored
	Transient Reactor Test Facility	18.3 m steel stack; 1.4 to 2.8 $m^3 s^{-1}$ discharge capacity; periodic cryogenic gas samples
	Zero Power Plutonium Reactor	22.9 m stack; 2.3 $m^3 s^{-1}$ discharge capacity; continuously monitored
	Central Facilities Area	Normal ventilation exhausts; onsite laundry; 12.4 $m^3 s^{-1}$ flow; continuously monitored
	Idaho Chemical Processing Plant	
	CFSGS	46 m stack; 42.9 $m^3 s^{-1}$ discharge rate; monitoring pending
	FAST	50 m stack; 47.2 $m^3 s^{-1}$ discharge continuously monitored
	Main Stack	76.2 m stack; 47.2 $m^3 s^{-1}$ discharge capacity; continuously monitored

Table A1. Major Release Points to the Environment from INEEL Facilities[a] (continued)

Type of Discharge	Area/Location	Type and Description of Discharge Structure
Airborne	Naval Reactor Facility	17.7 m discharge height; 16 to 22 $m^3 s^{-1}$ discharge capacity; continuously monitored
	A1W	
	A1W-RWDS	Near-surface discharge; 0.02 $m^3 s^{-1}$ discharge capacity; continuously monitored
	A1W & S5G	Cooling towers (2); capacity 1.3×10^6 L each
	Expended Core Facility	24.7 m stack; 35.4 $m^3 s^{-1}$ discharge capacity; continuously monitored
	Non plant areas	7.6 m stack; 0.3 $m^3 s^{-1}$ discharge capacity; continuously monitored
	S1W	44.8 m stack; 10.4 $m^3 s^{-1}$ discharge capacity; continuously monitored
	S5G	23.2 m stack; 8.0 $m^3 s^{-1}$ discharge capacity; continuously monitored
	Power Burst Facility	24.4 m stack; 1.9 $m^3 s^{-1}$ discharge capacity; continuously monitored Cooling tower; capacity of 2.95×10^{-6} L; maximum flow 57 liters per month
	Test Area North decontamination area	12.2 m stack; 20.8 $m^3 s^{-1}$ discharge capacity; continuously monitored
	Test Reactor Area	
	Advanced Test Reactor	76.2 m stack; 30.7 $m^3 s^{-1}$ discharge capacity; continuously monitored Cooling tower; 2.4 m × 66 m × 14.6 m. Evaporation rate at full reactor power 6.6×10^3 liters per month
	Engineering Test Reactor	76.2 m stack; 7.1 $m^3 s^{-1}$ discharge capacity; continuously monitored Cooling tower; 3 m × 113 m × 12.8 m. Evaporation rate at full reactor power 3.8×10^3 liters per month
	Materials Testing Reactor	76.2 m stack; 16 m $s^3 s^{-1}$ discharge capacity; continuously monitored

Table A1. Major Release Points to the Environment from INEEL Facilities[a] (continued)

Type of Discharge	Area/Location	Type and Description of Discharge Structure
Liquid (Injection Well)	WRRTF	94.5 meters deep; flow intermittent
Liquid (Seepage ponds; leaching pits; cribs)	Argonne National Lab-West Experimental Breeder Reactor-II	Batch monitored subsurface crib. Sanitary lagoon; 3 ponds (approx. 2.3 acres); flow rate 5.8×10^7 liters per year. Industrial pond; approx. 3 acres; flow rate 1.4×10^8 liters per year.
	Transient Reactor Test Facility	TREAT septic tank. Effluent to tile field. Flow rate approx. 5.5×10^6 liters per year
	Auxiliary Reactor Area	Surface depression (approx. 1/3 acre); estimated flow 1.05×10^7 liters per year; continuously monitored. 5 septic tanks; effluent for 2 tanks to underground tile field (ARA II). Effluent from 1 tank to surface depression approx. 1/3 acre (ARA I). Effluent from 2 tanks to surface depression approx. 1/2 acre (ARA III).
	Central Facilities Area	Sewage plant tile drain field; 610 × 61 m; average flow approximately 15×10^6 liters per month; continuously monitored
	Idaho Chemical Processing Plant	Percolation pond (77 m × 107 m × 3.7 m deep); inflow constantly monitored with a detection limit of 2×10^{-6} µCi per ml; flow approximately 2.3×10^8 liters per month. Sewage plant to tile field; annual flow 5.0×10^7 liters
	Loss-of-Fluid Test Facility	Continuously monitored pond with dimensions of approximately 76 × 152 × 5.5 m deep maximum. Septic tank. Effluent to the tile field. Annual flow approx. 2×10^6 liters.
	Naval Reactor Facility	Continuously monitored leaching beds handling 95,000 liters per month. 2 sewage ponds; 1.25 acres each. Total annual flow approximately 7.3×10^7 liters. Waste ditch; flow rate 7.0×10^8 liters per year.
	Power Burst Facility	Warm waste well; 0.25 m diameter × 33.5 meters deep; annual flow 7.8×10^6 liters.

Table A1. Major Release Points to the Environment from INEEL Facilities[a] (continued)

Type of Discharge	Area/Location	Type and Description of Discharge Structure
Liquid (Seepage ponds; leaching pits; cribs)		Evaporation pond; 45.7 m × 45.7 m × 1.7 m. Lines pond with capacity of 3.5×10^6 liters. Annual flow 1.2×10^6 liters.
	Test Area North	Continuously monitored pond approximately 13 hectares in size. Capacity 1.8×10^9 liters. Annual flow 9.7×10^7 liters
	Test Reactor Area	Two ponds: 40 × 73 meters and 76 × 122 meters; average flow approximately 90×10^6 L per month; continuously monitored. Chemical waste pond with dimensions of 52 × 52 × 1.5 meters. Capacity of 4.4×10^6 liters. Annual flow 6.6×10^7 liters. Sewage plant with leaching pond. Average flow 62 liters per minute.
Solid	All facilities	Low-level waste buried at Radioactive Waste Management Complex Transuranic waste stored at Radioactive Waste Management Complex High-level waste processed at ICPP Non hazardous wastes buried at INEL sanitary landfill; hazardous wastes shipped offsite

[a] Bowman et al. 1984