FINAL REPORT

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

Centers for Disease Control and Prevention
Department of Health and Human Services

October 8, 2002

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

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EXECUTIVE SUMMARY

The objective of this work performed by *Risk Assessment Corporation* (*RAC*) under Task Order 5 for the Centers for Disease Control and Prevention (CDC) was to use screening methods to rank the relative importance of potential historic exposures related to past releases of radioactive materials (radionuclides) to the environment from the Idaho National Engineering and Environmental Laboratory (INEEL) for the period 1952–1992. The ranking process was intended to focus future dose reconstruction efforts so that resources can be allocated to the radionuclides, time periods, and events that represent the most significant potential contributors to dose.

To evaluate INEEL releases and their potential to expose the public, we reviewed airborne, liquid, and solid radioactive waste disposal practices and discharge information. Waste materials were discharged into the environment from the INEEL, either directly to the air through stacks, or from accidents and special programs (episodic events), or to the groundwater when liquid wastes seeped into the ground. Surface streams or rivers do not flow from the INEEL Site to offsite locations so direct exposure through the surface water pathway is not realistic. However, some of the waste migrated into the groundwater either from the wells and ponds, or from underground solid waste at the radioactive waste storage areas at the INEEL. We studied the possibility of members of the public being exposed through contact with groundwater in the past and found this was not a significant historical exposure pathway.

Groundwater samples have been analyzed for more than 20 chemicals and radionuclides at various times since 1949. Of those analytes, tritium, ^{90}Sr, ^{60}Co, ^{137}Cs, ^{129}I, ^{238}Pu, ^{239,240}Pu, and ^{241}Am, have been detected in groundwater at onsite locations. In addition, tritium was detected in wells at the Site boundary during the years 1983, 1984, and 1985. A screening analysis for tritium in groundwater indicated that groundwater exposure pathways likely ranked low relative to routine releases of radionuclides to air for most years and relative to most episodic events. However, this study analyzed only offsite exposures to groundwater in the past and did not address the possibility of future exposure to the public via radionuclides in the groundwater. Evaluating current or potential future exposures to groundwater was beyond the scope of Task Order 5.

We determined that, in the past, radionuclides released to the atmosphere were the principle source of potential exposure to members of the public. 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 radioactivity. The release data measured since the mid-1970s provided more detail with regard to the specific radionuclides released.

Activities at the INEEL have resulted in routine operational releases that took place throughout the year and isolated episodic releases (short term releases from planned research programs or accidents) that took place only during a period of days or weeks. The Task Order 5 work involved identifying the facilities, programs, and episodic events that were the sources of important releases and the time periods during which these releases occurred. In ranking the relative importance of releases, we considered the quantities of specific radionuclides released, and the potential for these releases to be a source of exposure for members of the public at both onsite and offsite locations that represent likely points of maximum concentration, based on their

proximity and general relation to the facilities or episodic events from which the majority of releases occurred.

We evaluated routine operational releases separately from episodic releases because of the different assumptions and methodology used to assess their potential importance. For routine operational releases, we ranked the individual radionuclides released to the environment each year in terms of their potential importance to human health. The ranking process also identified specific years that were most important, the radionuclides that ranked highest for individual years, and the key facilities. For the episodic releases, we placed each release event into one of 5 different groupings, based on the timing of the release, existing meteorological conditions, and a number of other operational factors. We then ranked the events within each group. The goal was to create distinct groups of releases for which the assumptions required to characterize and evaluate the releases within each group were generally consistent to minimize the potential for introducing inconsistent biases that could significantly affect the ranking results. This approach does not allow the ranking values from one group of episodic releases to be directly compared to the ranking values within another group; however, some generalized comparisons are possible.

We used the screening methods developed by the National Council on Radiation Protection and Measurements (NCRP) to rank the INEEL releases. The NCRP methodology provides an efficient and conservative way to rank 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 importance of each release was also evaluated using the NCRP atmospheric screening factors.

To confirm that the NCRP method was suitable for these ranking purposes, we also made dose calculations using the Radiological Safety Analysis Code (RSAC), developed at the INEEL. The goal of these additional calculations was to demonstrate that the NCRP screening method was valid method for identifying the radionuclides and years that were most important in terms of public health. Duplicate calculations were carried out for several years of routine releases, and for various episodic events, and the ranking results were compared. In all cases, the RSAC code confirmed the ranking obtained using the NCRP methods.

To obtain the information needed to rank the INEEL releases, we used a variety of sources but focused primarily on Site documents reporting effluent monitoring procedures and data. The majority of historic monitoring and record keeping came from the Site 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. With this 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 needed to estimate quantities of radionuclides that were not measured during particular years of operations, or from particular facilities or episodic events.

To rank routine releases to air from the INEEL, we evaluated all pathways of exposure in the NCRP methodology and all individual years for 56 radionuclides released between 1952 and 1992. We assumed two different locations to assess potential exposure to routine releases: (1) an offsite location at Atomic City where all pathways of exposure were considered (This location is a likely point of maximum offsite concentration, based on its proximity and general relation to the facilities from which the majority of releases occurred); and (2) an onsite location near Highway

20, where the inhalation and plume immersion pathways are the important pathways to consider. The ranking values calculated for these locations were used to determine the relative importance of different radionuclides, time periods, and facilities.

The output from this evaluation process was a list of the radionuclides and their relative ranking values from all pathways of exposure for the offsite location or the inhalation and plume immersion pathways for the onsite location. We compiled the release estimates and ranking results of the NCRP screening methodology and the RSAC program in Excel spreadsheets. For radionuclides released to air from the INEEL, the highest ranked radionuclides at the offsite location, when all pathways were considered, were ^{137}Cs, ^{131}I, and ^{90}Sr, whether we considered all years of release together or focused on the high release years during the late 1950s. At the onsite location, where the inhalation and plume immersion pathways were most important, the highest ranked radionuclides included ^{41}Ar, some krypton and xenon isotopes, ^{144}Ce, ^{90}Sr, ^{106}Ru, and ^{131}I, for most of the high release years. Of the facilities at the INEEL, releases from the Idaho Chemical Processing Plant (ICPP) ranked highest during most early operational years after 1953. The Test Reactor Area (TRA) releases ranked highest during 1952 and 1953, the first 2 years of plant operation.

The INEEL had many episodic events that resulted in the release of radionuclides to the atmosphere over a relatively short period of time. These short duration events 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. Several episodic events resulted in the release of known quantities of specific radionuclides to the atmosphere. 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 to identify 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.

We assessed the potential importance of 99 individual release events by applying the NCRP air screening factors to an estimated average air concentration for each radionuclide comprising a given episodic release. Estimated air concentrations were based on dispersion factors specific to each release event wherever possible. Using the estimated air concentration for each radionuclide and the corresponding NCRP air screening factor, we calculated a ranking value for each radionuclide and exposure pathway. The sum of these ranking values represented the total ranking value used to evaluate the relative importance of each release event. As with the routine releases, a potential onsite and offsite exposure location was evaluated for each episodic release, and the onsite exposure scenario considered only the inhalation and immersion pathways. Consideration of the ingestion pathway for the offsite exposure scenario was dependent on the timing of the release (i.e., ingestion was not considered an important pathway for releases occurring during months outside the growing season).

Some of the episodic release events with high relative ranking values were the

- Initial Engine Test-10 (December 1957 through March 1958)
- Initial Engine Test-4 (April to June 1956)
- Fuel Element Burn Test B (1957)
- SL-1 reactor accident (January 1961)
- Ruthenium releases from the Waste Calcining Facility at the ICPP (1964)
- Fuel Element Cutting Facility filter break at the ICPP (October 30, 1956)

- Criticality accident at the ICPP (October 1959)
- RaLa releases (May 28, 1958; March 1, 1958; and October 7, 1957)

Because airborne releases from RaLa runs occurred for days to weeks after an operation over the course of 6 years, we included them as part of the routine releases. However, we also evaluated several RaLa runs that released significant amounts of material in a short time as episodic events.

In addition to routine releases and episodic release events, we evaluated hunting and eating waterfowl from the INEEL area as a special exposure scenario in response to concerns from some members of the community. We assessed three special exposure pathways concerning duck hunters: ingestion of meat, 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. We calculated ingestion doses from ^{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 TRA Ponds. Ingesting the duck meat was the most important exposure pathway, and ^{137}Cs was the largest contributor to the ingestion dose. While the potential significance of this special waterfowl pathway is likely less than many of the chronic or episodic releases, it may be an important exposure pathway for some individuals who are hunters in the INEEL region.

Our relative ranking approach identified some potential areas of consideration if additional resources and time were to be focused at the INEEL. We identified several episodic release events that ranked highest in terms of 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 one or more years during this time would allow for an assessment of health impacts associated with both routine annual releases and with shorter-term episodic releases.

CONTENTS

FIGURES

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 ratio
GRID III	Location of Fuel Element Burn Tests and other release tests
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 Test
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/PER	Power Burst Facility
RAC	*Risk Assessment Corporation*
RAL	Remote Analytical Laboratory
RaLa	Radioactive Lanthanum
RSAC	Radiological Safety Analysis Computer Program
RWMC	Radioactive Waste Management Complex
RWMIS	Radioactive Waste Management Information System
SL-1	Stationary Low-Power Reactor No. 1
SMC	Special Manufacturing Capability
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
WCF	Waste Calcining Facility
WERF	Waste Experimental Reduction Facility
ZPPR	Zero Power Plutonium Reactor (chenged to, Zero Power Physics Reactor)

GLOSSARY OF TERMS

Activation is the induction of radioactivity in material by irradiation with neutron radioactive material, a radiation-generating machine, or a nuclear reactor.

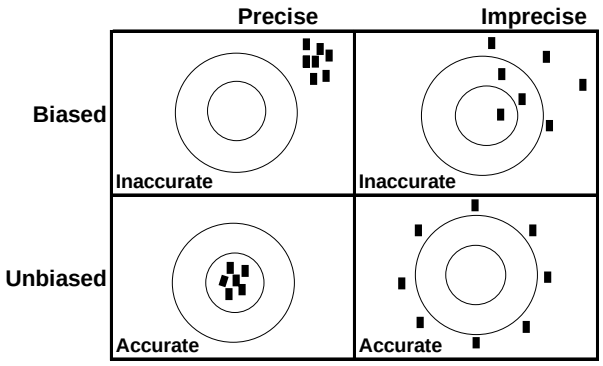
Activation products are radionuclides that result from the absorption of neutrons by uranium and other materials present in a nuclear reactor. An example is plutonium-239 produced following neutron absorption by uranium-238 and subsequent decays of uranium-239 to neptunium-239 and then to plutonium-239.

Activity is a measure of the rate at which a material is emitting nuclear radiation, usually given in terms of the number of nuclear disintegrations occurring in a given quantity of material over a unit of time.

Aquifer is a subsurface formation containing sufficient saturated permeable material to yield significant quantities of water.

Background radiation is the amount of ionizing radiation to which a person is exposed from natural sources, such as radiation from naturally occurring radionuclides in the soil, or cosmic radiation originating in outer space.

Bias is a systematic distortion of measurements that makes the results inaccurate. Accuracy is a measure of how close a value is to the true number or a measure of the correctness of a measurement. Precision is a measure of the exactness of a measurement.



Calibration is the use of environmental data collected under known conditions to represent model parameters outside the limits of the special conditions (for example, at other times and in other locations).

Chemical symbols are abbreviations for different elements and compounds. Examples of elements include U for uranium, O for oxygen, N for nitrogen and F for fluorine.

Computer code is a set of alphanumeric instructions that tells a computer to so something. A computer program consists of code.

Contamination refers to unwanted radioactive material or to the deposition of radioactive material in the environment or in any place where it may make surfaces or equipment unsuitable for some specific use.

Criticality is the condition in which a material undergoes nuclear fission spontaneously; the critical mass of a material is that amount that will self-sustain nuclear fission when placed in an optimum arrangement.

Decay (daughter) products refer to the isotopes or radionuclides that result from radioactive decay of isotopes, such as the uranium and thorium isotopes.

Deposition velocity refers to the proportion of the rate of deposition of radioactive material on the ground to the concentration of the material in the air at a specified reference height. The dry deposition velocity is a function of the particle size and density of the radioactive material (the larger the particle size, the greater the deposition velocity); the nature of the

surface (for example, snow-covered, lawn, tree-covered); and meteorological variables (for example, the higher the wind speed, the higher the deposition velocity).

Direct radiation exposure refers to one pathway of exposure of people to radiation from the INEEL. In this exposure pathway, penetrating radiation emitted from radioactive material is partially absorbed by individuals exposed to it. The amount of exposure decreases with distance from the source.

Dose is a general term denoting the quantity of radiation or energy that is absorbed by the body. There are technical terms with specific definitions, such as absorbed dose, equivalent dose, and effective dose.

Dose Reconstruction is a scientific study that estimates doses to people from releases of radioactivity or other contaminants into the environment from a facility.

Effluents are gases or liquids containing contaminants that flow from a process, building, or site into the surrounding environment.

Environmental exposure is exposure to radiation through environmental pathways.

Episodic releases include both accidental and intentional releases that occurred over short periods of time such that they may require special treatment during environmental transport and dose assessment.

Excursion is a sudden rapid increase of power produced when a reactor or other system of fissile material undergoes a sudden increase in reactivity.

Exposure pathways are ways in which people are exposed to contaminants in the environment. The key exposure pathways are air and water, with most exposures occurring by inhalation, drinking water, eating crops and other foods, and from direct irradiation.

Extraction is a chemical process for selectively removing materials from solution.

Fissile refers to a material capable of undergoing fission by any process.

Fission products are radionuclides that result from the splitting of heavy elements like uranium in a nuclear reactor. Examples of fission products are ^{90}Sr, ^{99}Tc, ^{106}Ru and ^{137}Cs.

Fission product inventory refers to the quantities and types of radionuclides resulting from a nuclear reaction (e.g., reactor operation or criticality), as in a fuel element.

Fuel (nuclear, reactor) refers to fissionable material used as the source of power when placed in a critical arrangement in a nuclear reactor.

Fuel reprocessing refers to the chemical processing of irradiated (spent) nuclear reactor fuel to recover useful materials as separate products, usually separation into plutonium, uranium, and fission products.

Gaussian plume model is a well-known air transport model that assumes that released materials are moved in a straight line from the source, depending upon the wind speed and direction at the time of the release.

Health impact is the likelihood of deleterious health effects occurring as the result of exposure.

Ionizing radiation is a type of radiation that has enough energy to create ions (ionized atoms that are chemically active) inside living cells. These ions can damage key substances in cells, including the DNA within the cell nucleus.

kilo is a prefix that multiplies a basic unit by 1000. For example, 1 kilogram = 1000 grams.

Mathematical model is a collection of mathematical formulas used to characterize a relationship or process.

Median is the central point of a distribution. Half of the values are larger than the median value and half are smaller.

pico is a prefix that multiplies a basic unit by 1/1,000,000,000,000 or 1×10^{-12}. For example, 1 picocurie equals 1×10^{-12} curie.

Radionuclide is a radioactive isotope of a particular element, for example ^{137}Cs or ^{238}U.

Receptor location is a geographic location of individuals where concentrations are calculated by a model.

Risk is the probability of a deleterious health effect, such as cancer, being induced by radiation.

Rupture is a breach of the metal cladding of a production rector fuel element thereby releasing radioactive materials to reactor cooling streams.

Screening is the process of rapidly identifying potentially important radinuclides, chemicals, and exposure pathways by eliminating those of known lesser significance; that is, a process of evaluating releases of radionuclides or chemicals from a site or facility based on very conservative assumptions about release estimates and exposure parameters so that detailed studies can be reserved for the more important releases that affect human health.

Screening factor is a term used by the National Council of Radiation Protection and Measurements (NCRP 1996)) to represent the sum of committed effective doses (Sv) received from external radiation over an assumed time period (less than 1 y) plus inhalation of radioactive particles and gases, plus ingestion of locally grown foods during 1 y for a unit concentration of radioactivity (Sv Bq^{-1} m^3) in the medium, atmosphere or water.

Screening value is the value that results after considering the screening factor, environmental transport, uptake and metabolism of a particular radionuclide. While the screening value is reported in units of dose (millirem or sievert), it does not represent a true dose because of many conservative assumptions used in the screening analysis for both routine and episodic releases. NCRP (1996) emphasizes that the "doses" (or screening values) estimated by screening techniques "are strictly for comparison with an environmental standard (limiting value) and are not intended to represent estimates of actual doses to individuals. Furthermore, the endpoint of the screening assessment is a reference individual as opposed to a real person."

Seepage pond is an artificial body of surface water formed by the discharge from an industrial process, used at the INEEL for liquid process discharge.

Short-lived isotope is a radionuclide that decays so rapidly that a given quantity is transformed into its daughter or decay products within a short period

Source term refers to the quantity, chemical and physical form, and the time history of contaminants released to the environment from a facility.

INTRODUCTION

The objective of Task Order 5 for the Centers for Disease Control and Prevention (CDC) was to use screening methods to evaluate historic releases of radioactive materials (radionuclides) to the environment from the Idaho National Engineering and Environmental Laboratory (INEEL) during the period 1952-1992. Screening refers to the process that identifies potentially significant radionuclides, episodic releases from special programs or accidents, and/or exposure pathways. This process provides an efficient way to rank the various radionuclide releases to the environment in terms of their potential impact on human health so that, if warranted, further in-depth study of certain radionuclides, years, or events can focus on the most important releases.

Reconstructing releases of radioactive materials from a facility like the INEEL can be a time-consuming and resource-intensive process. 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 intentional release 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 reviewed airborne, liquid, and solid radioactive waste disposal practices and discharge information at the site to determine the most likely ways members of the public might have been exposed. Radionuclide releases to air were identified as the most significant source of historical exposures. We evaluated the potential for exposure to individuals at an onsite and offsite location. We based the relative ranking on a screening or ranking value[1] calculated with the screening methods developed by the National Council on Radiation Protection and Measurements (NCRP) (NCRP 1996). The NCRP methodology provides an efficient and conservative way to prioritize radionuclides released over the long term so that resources can be focused on the most important releases.

The INEEL had many episodic events that resulted in the release of radionuclides to the atmosphere over a relatively short period of time. For these episodic events, we also evaluated the importance of each release by applying the atmospheric screening factors developed by the NCRP (NCRP 1996). Several episodic events resulted in the release of known quantities of specific radionuclides to the atmosphere. 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 importance of each release. While the NCRP screening factors and models were not designed to assess episodic or short-term releases because of underlying assumptions within the model, our results indicate that the screening factors are viable tools for ranking and prioritizing episodic or short-duration releases.

[1] The ranking value is the value that results after considering the screening factor, environmental transport, uptake, and metabolism of a particular radionuclide.

As a comparison with the NCRP methods, we evaluated a number of years of chronic releases and qseveral episodic release events by calculating doses using the Radiological Safety Analysis Code (RSAC) developed at the INEEL and risk coefficients from the U.S. Environmental Protection Agency (EPA) (Eckerman et al. 1999). The comparisons are important to confirm or refute the efficacy of the NCRP screening methods for ranking the relative importance of chronic releases and release events. The NCRP and RSAC results agreed very well confirming that the NCRP methodology is an acceptable method to rank releases of radionuclides, years of release, separate facilities, and episodic events from a site like the INEEL.

This report presents the ranking results of the screening process for routine and episodic releases. To provide some background and perspective, we initially prsenet a brief overview of the history of some of the INEEL facilities before describing our screening methods. Additional sections within the report present results of our analysis of the possibility of members of the public being exposed to radionuclides through contact with groundwater in the past. We found this was not a likely historical exposure pathway. However, our analysis only evaluated potential offsite exposures to groundwater in the past and did not make any judgments regarding onsite exposures or future offsite exposures. We also evaluated hunting and eating waterfowl from the INEEL area as a special exposure pathway in response to concerns from some members of the community. While the potential significance of this special waterfowl pathway is likely less than many of the chronic or episodic releases, it may be an important exposure pathway for some in the INEEL region.

This study identified several important areas to consider if the CDC were to focus additional resources at the INEEL. The CDC, INEEL Health Effects Subcommittee (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.

STUDY BACKGROUND

In August 1991, DOE published a historical radiation dose evaluation (HDE) 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 HES, a Federal advisory committee, whose members provide advice to CDC on community concerns about CDC's dose reconstruction activities at INEEL. In 1994, as part of the first phase of work, the CDC developed a database of documents relevant to an environmental dose reconstruction at the INEEL (SC&A 1994).

In 1997, the CDC funded *Risk Assessment Corporation* (*RAC)* to undertake several smaller and more focused studies on the feasibility of completing a more detailed dose reconstruction at the INEEL. These smaller projects, called task orders (Task Orders 1, 5, and 6), focused on reviewing the documents in the Phase I database, and any other relevant documents to catalog INEEL releases of chemicals (Task Order 1) and radionuclides (Task Order 5). Based on information from these historic documents, *RAC* was to evaluate these releases in a timely manner, and list in order by priority those releases and/or radionuclides that may warrant further detailed 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).

As the document review phase for work on Task Order 5 proceeded, it became clear that documents in a number of the boxes at the INEEL had not been evaluated in 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. The INEEL document database is available at the CDC website, http://www2.cdc.gov/nceh/Radiation/INEEL/dev.htm. It has not been a part of the scope of this task to change the structure and function of the original Phase I database.

For Task Order 5, we ranked historical exposure to past releases of radionuclides from the INEEL so that future attention could be focused on those releases, facilities, or episodic events that appeared 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 radionuclides that were released than for chemicals. In addition, various historical studies have already evaluated many specific radionuclide release events, and government programs at the INEEL.

BACKGROUND ON INEEL SITE AND FACILITIES

The INEEL is unique among the DOE facilities because it is a large complex site with many independent contractors, goals, and missions. Our approach for 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 were the largest contributors to radionuclide releases. In this section, we provide a brief overview of the history and layout of some INEEL facilities. These facilities are briefly described with regard to their function and purpose, and the air and liquid effluent monitoring systems that were in place. We used the total annual reported radionuclide releases from the Site to air and via liquid effluents to identify the time periods during which the largest releases occurred, and to ascertain the facilities and processes that contributed the most to these releases. Releases associated with specific episodic events are discussed individually in a later section of the report.

In 1949, the U.S. Atomic Energy Commission (AEC) established the National Reactor Testing Station (NRTS) in Idaho as a government site to build, test, and operate nuclear reactors. The Site also utilized a variety of support facilities and equipment. In 1974, the NRTS was renamed as 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 INEEL has operated 52 reactors plus fuel handling and reprocessing and radioactive waste

storage facilities since it began operations. By 1988, 13 reactors were still operable. Figure 1[2] shows the major facilities and their locations at the INEEL.

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. There are many small communities in the area with Idaho Falls the largest city within 50 miles of the Site (Figure 21). The population within a 50-mile radius of the operational center of the Site was approximately 70,000 in 1970 (ERDA 1977), about 120,000 people in 1990 (Hoff et al. 1992) and 121,500 people in 1995 (Mitchell et al. 1996). In 1974, about 60% of INEEL was open to cattle or sheep grazing. The grazing permits were administered through the Bureau of Land Management (BLM). No dairy operations used INEEL for grazing; most dairy cattle in local areas were maintained on stored feed (ERDA 1977).

Activities at the INEEL have resulted in chronic operational releases taking place throughout the year and 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). In addition to the information provided in this report about specific facilities and programs, the reader is encouraged to refer to the various references cited throughout this report for further details. 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 had to be reconstructed. This was also true for many of the operational releases occurring during this time period.

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 (NOAA) perform groundwater and meteorological monitoring, respectively. The radionuclide release data measured since the mid-1970s are more detailed with regard to the specific radionuclides released.. Before that time, particularly before 1962, much of the reported release data were in the form of gross activity only, which cannot be used directly for prioritizing radionuclide releases from the Site. There are additional sources of effluent monitoring data available in weekly and monthly reports, particularly from the Idaho Chemical Processing Plant (ICPP) (for example, see Hayden 1957-1963; Rich 1962; Williamson 1977). Scientists conducted an historical dose evaluation (HDE) in 1991 and identified apparent technical errors with some of the release data (DOE 1991a). These technical errors necessitated a careful review of historic reported and a recalculation of releases for establishing their relative importance. 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. Hogg et al. (1971a) provides additional information on the potential sources of radioactive airborne effluents from the ICPP, the Test Reactor Area (TRA), the Naval Reactors Facility (NRF), Argonne National Laboratory-West (ANL-W), the Special Power Excursion Reactor Tests (SPERT) conducted at the Power Burst Facility (PBF), the Auxiliary Reactor Area (ARA), and the Central Facilities Area (CFA).

[2] 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.

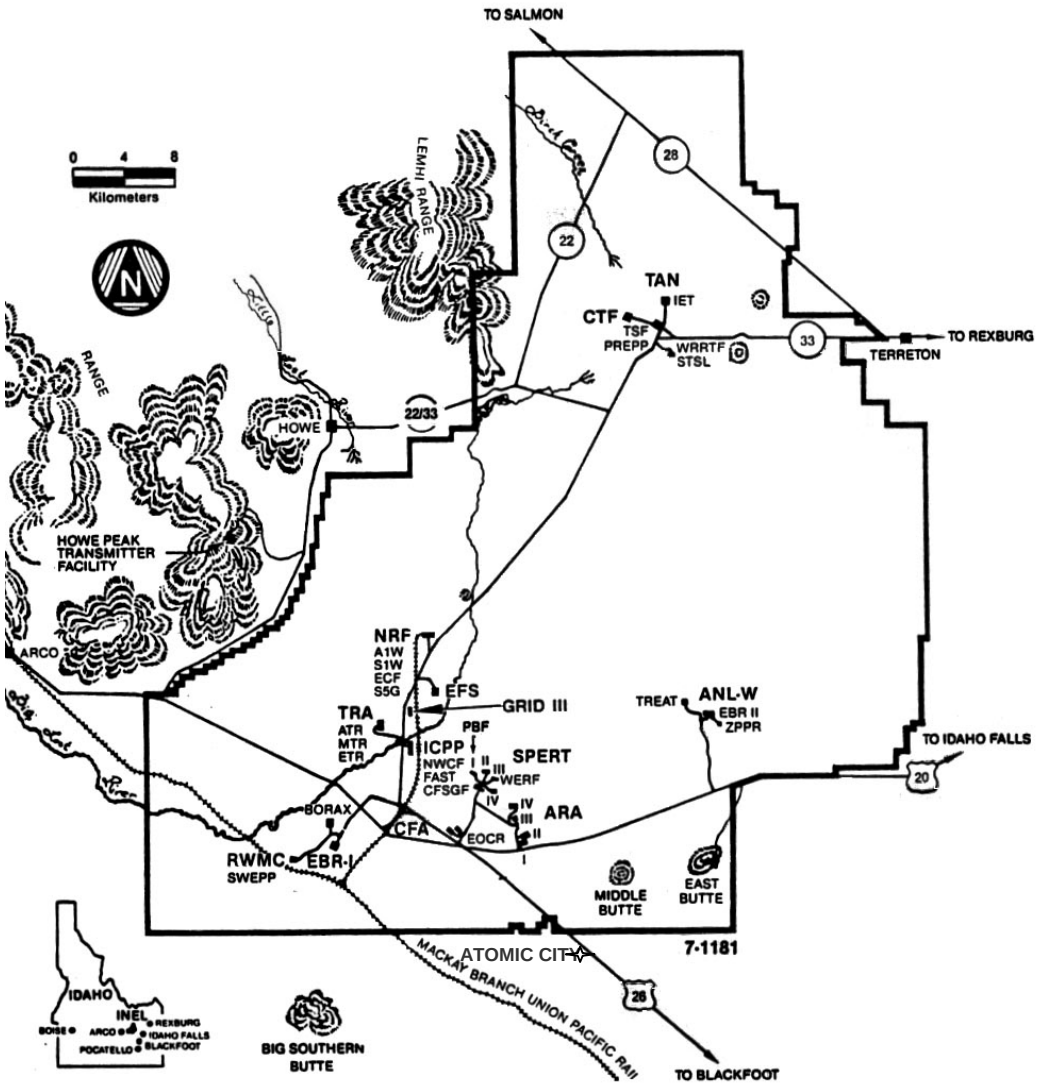


Figure 1. Map of the Idaho National Engineering and Environmental Laboratory. Facilities include the Argonne National Laboratory-West (ANL-W) where the Experimental Breeder Reactor No. 2 (EBR-II), Transient Reactor Test Facility (TREAT) and Zero Power Plutonium Reactor (ZPPR)[3] are located, Auxiliary 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.

[3] Later changed to Zero Power Physics Reactor.

Figure 2 provides a perspective on the total releases of radioactivity to air from the INEEL facilities. 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 ranking purposes, this temporal distribution of releases provides some perspective for understanding the potential significance of releases from various facilities and areas of operation onsite. Between 1952 and 1974, reported cumulative releases of total activity from the Site were 12 million Ci (DOE 1991a; ERDA 1977).

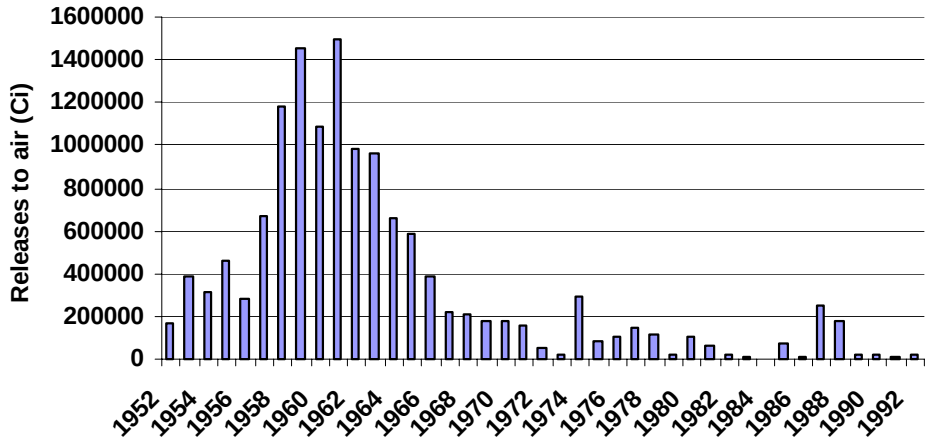


Figure 2. Total annual releases of radioactivity to air from routine releases at INEEL facilities (DOE 1991a; Hoff et al. 1992; Hoff et al. 1993).

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 figure illustrates the relative percentages of total discharges from the ICPP and TRA, which have accounted for greater than 95% of the total airborne radioactivity for all years except 1984-86. Releases from the Argonne National Laboratory-West (ANL-W) amounted to 5% in 1984 and 6% in 1986, and releases from the Loss of Fluid Test (LOFT) facilities amounted to 11% in 1985. TRA releases dominated until the start of the Radioactive Lanthanum (RaLa) project that was carried out at the ICPP between 1957 and 1963. Total radionuclide releases from ICPP then tailed off through about 1968, after which facility releases fluctuated and were dominated by both the TRA and ICPP. The ICPP contributed about 8 million Ci, and the TRA about 5.5 million Ci to the total reported Site releases from early 1953 through the late 1980s. In comparison, the Test Area North (TAN) and ANL-W facilities contributed only about 30,000 Ci each.

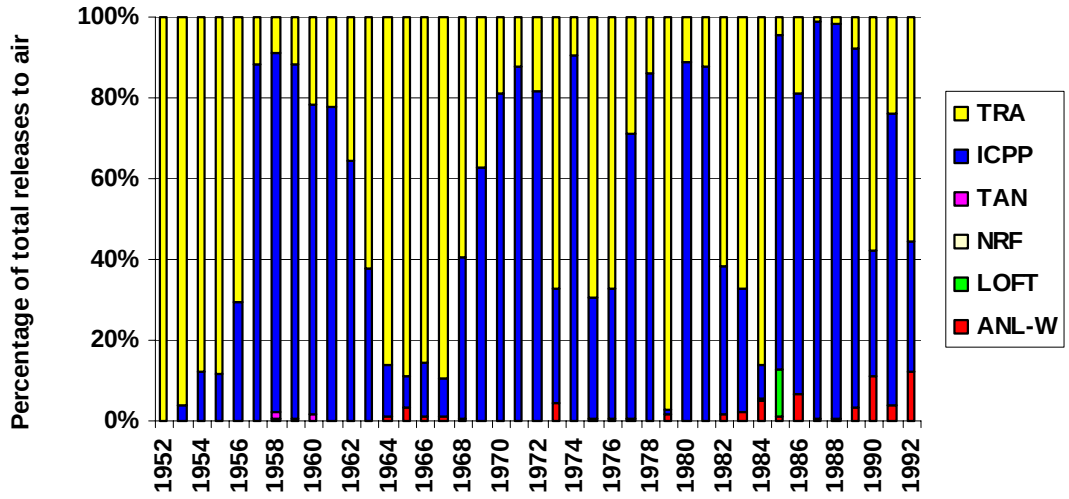


Figure 3. Percentage of total annual releases of radioactivity to air from 1952 through 1988 for various INEEL facilities and programs onsite (DOE 1991a).

The ICPP and TRA facilities also released the majority of liquid effluents. Liquid wastes were discharged to injection wells[4], seepage basins or pits, or seepage ponds depending upon the facility, and were generally monitored at the time of release. The INEEL Site has no surface streams or rivers flowing from onsite to offsite locations. 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. Except for the early 1960s, however, the TRA contributed over 80% of the radioactivity discharged from the INEEL in liquid effluents.

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 MTR began operations on March 31, 1952, at a power level of 30 MW(t); it provided fuel for a special program, the Radioactive Lanthanum (RaLa) Program, from 1957 to 1963 at ICPP. The MTR played an important role during the early years because of its design and its normal operations and experiences with cladding failures and other incidents are fairly well documented (De Boisblanc 1958).
- The ETR startup occurred in 1957 with an operating power level of 175 MW(t).
- The ATR was the world's largest test reactor when it began operations in July 1967.

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

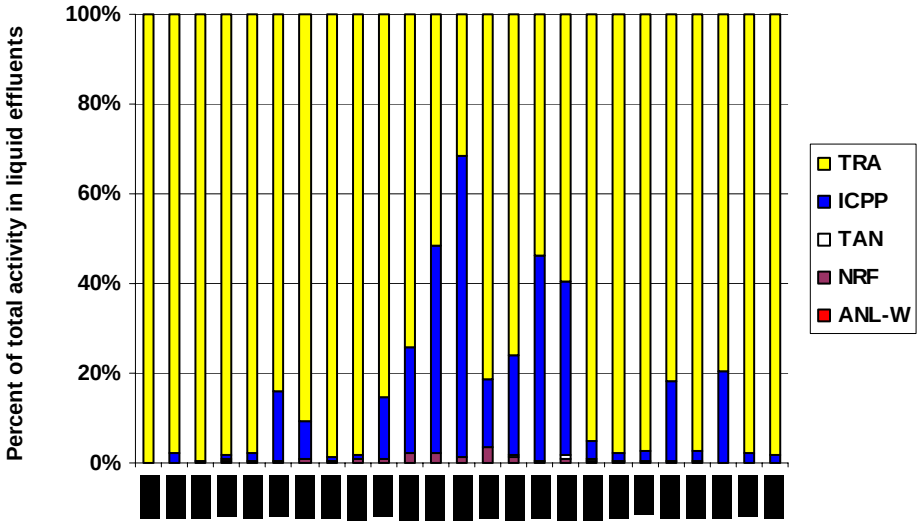


Figure 4. Percentage of total annual releases of radioactivity in liquid wastes from five facilities and programs at the INEEL from 1952 to 1974 (ERDA 1977; Osloond 1970). The Test Reactor Area was the largest contributor to radioactivity in liquid effluents for most years. For releases of liquid wastes in the early years, discharges were made to wells, seepage basins or pits, or seepage ponds depending upon the facility.

The ventilation system for the reactors was designed to use outside air, first brought through office areas, then to slightly contaminated areas, and finally into high radiation areas. Negative pressure was maintained in contaminated and high radiation areas (Hogg et al. 1971b). From these high radiation areas, the air was filtered and discharged to one of three 250-foot (76.2 m) stacks with monitors for gross alpha and beta activities, a gamma detector, and a charcoal filter for iodine collection (Bowman et al. 1984, Hogg et al. 1971b). From 1952–1974, the TRA reported airborne releases of just over 5 million Ci. This level can be compared with the total reported radioactive releases from the INEEL of about 12 million Ci during this same time period.

The TRA contributed the highest levels of radioactivity in liquid wastes, although the ICPP discharged the greatest volumes of liquid wastes. The major radionuclide 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 the 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; and hot wastes, which were too radioactive for immediate disposal to the groundwater.

Figure 5. Test Reactor Area (TRA) at the INEEL where the Materials Testing Reactor, the Engineering Testing Reactor and the Advanced Testing Reactor are located. Stacks from the three areas were all 250-ft (76.2-m) high.

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. Aerial view of the Idaho Chemical Processing Plant (ICPP) 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 250-ft (76.2-m) 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 bins; the ICPP processing corridors for fuel reprocessing; and the High Level Waste Tank Farm with 11 underground liquid storage tanks.

Airborne Releases

Airborne releases of radioactivity from the ICPP came primarily from the process dissolvers, waste solvent burner, analytical facilities, and the WCF, where calcination conditions determined the size and concentration of solids in the effluent. Various off-gas studies at the ICPP provided data on the operations of the effluent treatment systems (Wheeler 1959; Cederberg and Bower 1959). There were three off-gas streams, each with 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, including the airborne effluents from the WCF, were discharged through the main 250-ft (76 m) stack to the environment. The liquid waste storage tanks were also vented to this discharge air stream. While some airborne releases occurred from the solids

storage bins and through sample handling operations, these sources contributed negligible amounts (Lakey et al. 1963). In the early 1980s, the New Waste Calcining Facility (NWCF) came online and discharged airborne effluents from its own 72 ft (22 m) stack, with a discharge capacity of 51.9 m^3 (DOE 1982). There is extensive and detailed information available on the ICPP airborne waste treatment systems (Wheeler 1959; Cederberg and Bower 1959; Ayers and Burn 1960; Wong and Roberts 1979).

Figure 7 shows that the reported airborne releases from the ICPP between 1953 and 1974 was 7 million Ci with 6 million Ci released between 1957-1963 (ERDA 1977). Annual releases exceeded 500,000 Ci from 1957 through 1962, with peaks of 1.3 million Ci in 1959 and 1.1 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 during this period (see Figure 3).

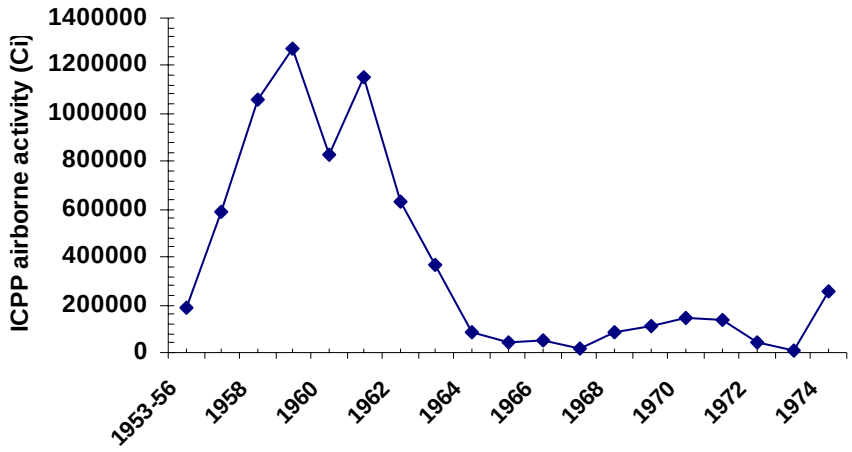


Figure 7. Annual releases to air from the ICPP from 1953–1974, a time period that included releases from the RaLa program (ERDA 1977).

While actual measurements of all radionuclides were not made on the airborne effluent, the Site calculated the release rates of various fission products from operational data. Beginning in 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.

The ICPP carried out 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 Site estimated release rates of ruthenium from pilot plant data (Lakey et al. 1963). The ICPP reported annual releases, at least for some years, of ^{90}Sr, ^{106}Ru, ^{137}Cs, ^{144}Ce, ^{147}Pm, and ^{239}Pu, ^{237}Np, ^{60}Co, ^{95}Zr, ^{89}Sr, ^{91}Y, ^{95}Nb, ^{103}Ru, and ^{141}Ce based on 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 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 HEPA filters.

Liquid Effleuent Releases

Liquid waste streams were generated from all areas of the ICPP and varied in volume and degree of contamination. A continuous liquid waste monitor was installed in 1954 in the ICPP (King 1956). The high level waste streams were collected in the tank farm before solidification in the WCF. All other radioactive liquid wastes were collected, processed, and discharged to the injection disposal well. Two intermediate and low-level liquid waste streams were collected in the evaporator tank before being discharged to the injection well. 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 (Dickey et al. 1972). Additional liquid wastes totaling about 15,000 gallons per year from other areas at the INEEL were handled in this system as well. The most abundant radionuclides measured in liquid waste discharged to the injection disposal well were tritium, ^{137}Cs, ^{106}Ru, and ^{90}Sr.

Test Area North

Test Area North (TAN) (Figure 8) was built in 1954 to support the U.S. Air Force Aircraft Nuclear Propulsion (ANP) Program to test nuclear engine prototypes and investigate the feasibility of a nuclear or chemical propulsion system for military use. This program conducted numerous Initial Engine Tests (IETs) during the late 1950s and early 1960s when the program was cancelled. The Technical Support Facility (TSF) provided support for the area. The only reactor operations at the TAN complex occurred at the Loss of Fluid Test (LOFT) facility. These programs are evaluated and discussed in the Episodic Release section of this report.

Figure 8. Aerial view of a portion of the Test Area North (TAN), location of the Aircraft Nuclear Propulsion Program and the Loss of Fluid Test Facility.

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 were discharged from a 168 ft (51-m) stack above grade (ERDA 1977). Operational releases from the TAN totaled approximately 54,000 Ci from 1958-1974, most attributed to the ANP Program and IETs. This value (54,000 Ci) can be compared to total reported releases from the INEEL of 10 million Ci during this same time. 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 to operate three major reactors: the 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. ANL-W is has the Hot Fuel Examination Facility (HFEF) and a 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-ft (61-m) 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-ft stack. The flow through the stack averaged 70,000 cubic ft per minute and all discharged air from the stack was monitored (Hogg et al. 1971b). All airborne effluents from the fuel assembly and storage building were treated as radioactive. The principal radionuclides identified in the stack effluent were tritium, ^{41}Ar, ^{85}Kr, and ^{133}Xe.

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 effluents from the reactor were exhausted through a bank of six parallel HEPA filters and discharged into a 60-ft (18-m) 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 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 (ERDA 1977).

Airborne releases of radioactivity were highest from 1965 through 1969 because the Fuel Cycle Facility at the ANL-W that processed EBR-II fuel lacked an adequate holdup system to reduce the release of short-lived radionuclides like ^{133}Xe and ^{135}Xe (Hogg et al. 1971b).

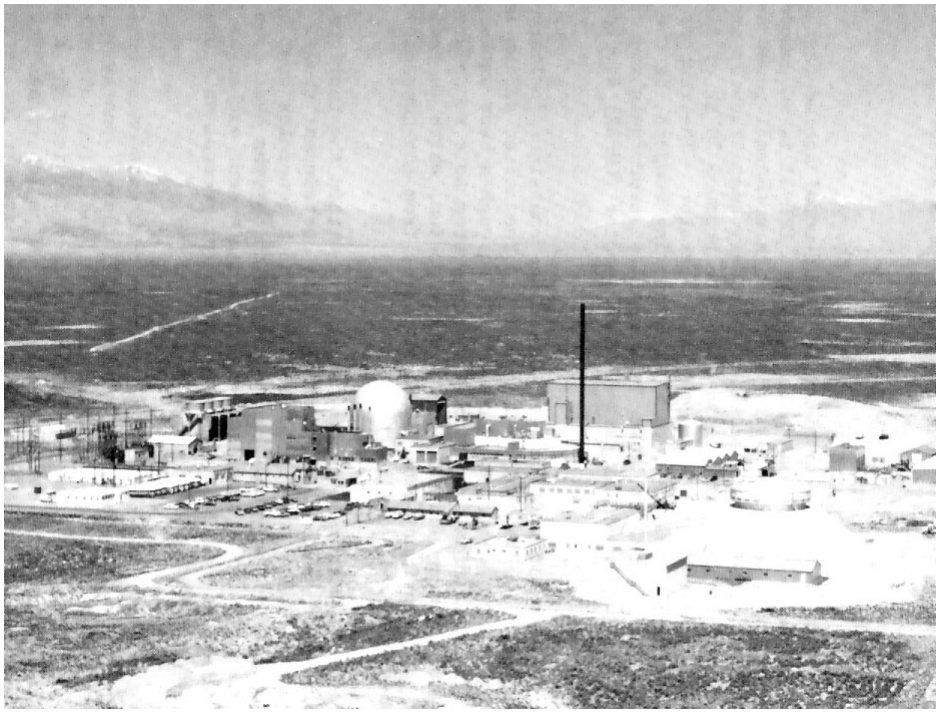


Figure 9. Aerial view of Argonne National Laboratory-West (ANL-W). The 200-ft (61-m) stack is visible to the right of center.

Liquid radioactive wastes form the ANL-W came primarily from the EBR-II Area. Low level radioactive liquid waste were piped to a 35,000-gallon underground seepage pit located outside the fenced area of the EBR-II Site. In 1973, the seepage pit was covered with 8 inches of concrete (Hogg et al. 1971b). 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 seepage pit in 1970.

Intermediate liquid radioactive wastes were transported through underground pipes or by portable tanks to retention tanks in the laboratory building waste evaporator at the ANL-W 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 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 ft) 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 about 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 beginning in 1966.

The Expended Core Facility (ECF), opened at the NRF in 1958, examined and tested components that had been irradiated in nuclear reactors and structural materials removed from expended naval core fuel modules (ERDA 1977). After separation, the structural parts were transported to the RWMC, 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 airborne effluents 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 facilities at the NRF. The liquid wastes were discharged to two fenced seepage basins 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 about 350 Ci (ERDA 1977). The major radionuclides in liquid wastes released to seepage basins from the NRF were ^{3}H, ^{14}C, and ^{60}Co. Releases were highest from 1958 through 1964 (Hogg et al. 1971b). 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. Until the late 1970s, the area of interest as a potential radioactive release source was the laundry. Laundering radioactively contaminated clothing generated about 340,000 gallons of liquid waste per month. 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 boiled in the reactor core and passed saturated steam directly to the turbine

for power generation. The BORAX-I test reactor operated from 1953-1954 and the BORAX-I excursion 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 began in 1955; the BORAX-IV reactor operated from 1956-1958, and 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, evaluated as an episodic event.

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. 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 program was a special government effort during the late 1940s and early 1950s to increase production of high specific activity barium, which decays to radioactive ^{140}La, an intense radiation source. An outmoded ^{140}Ba production facility at Oak Ridge National Laboratory (ORNL) in Tennessee, which had operated from 1945-1956, needed to be replaced. A search for a new production facility location led to the INEEL because of the availability of high specific activity fuel from the new MTR. The ORNL facility had achieved a maximum production of 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 the 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, the 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) building the Pilot Plant, (2) testing with unirradiated MTR fuel, and (3) active testing with irradiated fuel.

In less than two years, (November 1, 1955 to June 30, 1957), a startup program was conducted from plant takeover to successful high activity level production at the ICPP. During the document search phase of the Task Order 5 work, we located the original handwritten logbooks for the RaLa Program with daily entries from 11/29/53 through 1/5/68. These logbooks listed ^{131}I releases, several pages of calculations, and stack flow rates for each RaLa Program "run". The monthly ^{131}I releases to the atmosphere varied widely. The logbooks also recorded ICPP stack releases and liquid effluent releases to injection wells. The results of the RaLa Pilot Plant testing, laboratory research and development of the process for separating ^{140}Ba from the MTR fuel elements, and regular exchanges about the process are well documented (Anderson and Weech 1954; Anderson et al. 1959).

Because ^{140}Ba has a fairly short half-life of 12.8 days, the fuel was processed as soon as possible (usually within 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 short-lived gaseous fission products. For the RaLa program, however, fuel was reprocessed as soon as possible after removal from the reactor (normally 2-7 days). As a result, these RaLa runs resulted in large releases of fission products to air, with ^{131}I ($T_{1/2}$ = ~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 ($T_{1/2}$ = ~78 hours) decayed to ^{132}I ($T_{1/2}$ = 2.3 hours), which contributed much of the activity several days after processing. The $^{132}I/^{131}I$ ratio 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 quarterly technical progress 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 (1) combining with the sodium hydroxide in the caustic scrubber solution to form sodium iodide, or (2) collect in a 10,000 ft^3 gas holder and held for release "under favorable weather conditions"(Stevenson 1957). Neither of these two 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 inability to remove iodine-bearing dusts and particulates.

Hanford's Role in the RaLa Program

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 in this program at the INEEL. The historic record clearly shows that Hanford shipped fuel slugs (different configurations of fissionable material used as the source of power when placed in a critical arrangement in a nuclear reactor) 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 " …the GE Company (Hanford) believes that they will be able to undertake this service without too much difficulty …" (Hanford 1956-1958: HAN 42692). Shipments of 60-day cooled slugs were to begin in March 1954 according to a memo, dated 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 7½ kg per month...." (Hanford 1956-1958). These records indicate that fuel was indeed shipped to the INEEL, but after it had been cooled for weeks or months. Fuel for the RaLa runs was cooled at the most for about 2 days to limit the decay of the ^{140}Ba. The historic doucments 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 Hanford 1956-1958: HAN-61940.
[b] From Hanford 1956-1958: HAN-68946.

Documentation of RaLa Releases

Daily releases of ^{131}I and beta-minus iodine activity were reported beginning on July 7, 1957 (Hayden 1957-1963). 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 routine operational releases or as episodic events.

The daily release records for ^{131}I from the RaLa runs (Hayden 1957-1963), in general, do support those summarized in DOE (1991a) with 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). Another small difference is seen when 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. Overall, documentation of releases from the RaLa runs 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). There is a considerable decline in ^{131}I releases after August 1958, when charcoal beds were installed to remove iodine from the airborne effluent. When the weekly ^{131}I releases in 1957 and 1958 from the ICPP are examined, it points out the difficulty of completely separating out the releases from the RaLa operations from other processing activities at the ICPP during that time (Hayden 1957-1963).

Plotting the daily releases from October 1, 1957 through mid-February 1958 illustrates that ^{131}I releases could occur for up to several days after the completion of a RaLa run (Figure 10). For example, the highest releases associated with RaLa run #7 occurred on the day of the run (~41 Ci). However, for RaLa run #8, which occurred on October 21, 1957, the highest releases following that run occurred two days later, on October 23 (~26 Ci). Approximately 3.5 Ci were released on the day of RaLa run #9, but over 15 Ci were 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 associated with the RaLa runs as routine operational ICPP releases in our ranking process. However, several of the RaLa runs conducted before the charcoal beds were installed in August 1958 released significant quantities of iodine to the atmosphere (see Tables 37 and 40). Some of these latter RaLa runs were also evaluated as episodic events in the RaLa Iodine Releases section.

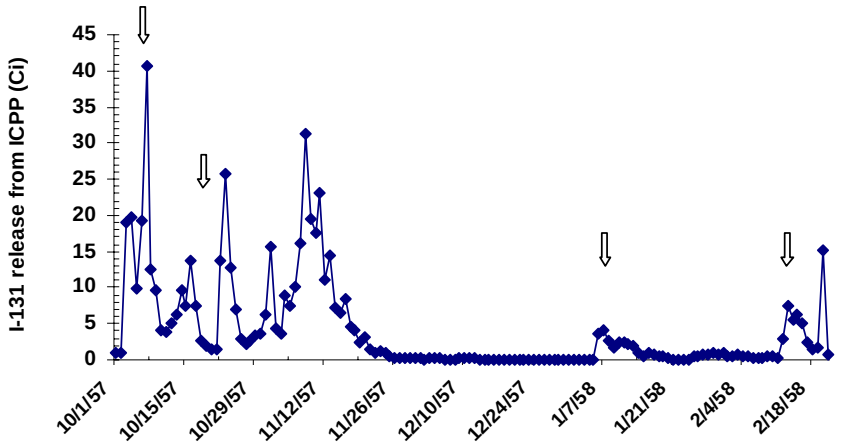


Figure 10. Daily releases of ^{131}I from the ICPP from October 1, 1957 through February 15, 1958. The block arrows indicate the occurrence of RaLa runs #7 (October 7, 1957), #8 (October 21, 1957), #9 (January 6, 1958), and #10 (February 15, 1958).

EVALUATION OF ROUTINE RELEASES

We used a wide range of available documents and reports to obtain source term estimates for for the routine releases ranking process to determine the INEEL facilities, radionuclides, and years that may have been most significant for human health. These documents included the INEEL 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, 1980, 1983; Hoff et al. 1984); and numerous 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 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 (Table 2). This exercise confirmed 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) provided a solid foundation for our ranking process. For example, Table 2 compares the annual reported release of ^{131}I and ^{132}I from data compiled from various types of reports.

Agreement among daily, monthly, and annually reported releases for various radionuclides 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 1983 (ENICO 1983b). In the final tally of releases of ^{238}Pu to air through the main stack at the ICPP, the annual total was an order of magnitude lower than the total obtained from summing the monthly totals. Combining the monthly release estimates yielded an annual total of 1.45 millicuries (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 in 1983. Instead, 63.2 mCi should have been reported (WINCO 1984). Transposing the value for the 1st quarter total caused this error; the reported total of 21.6 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 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.

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[b]	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.

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 of 1983 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 (ENICO 1983b).

Toward the end of 1984, an ambient air monitoring program for particulate matter was initiated at the ICPP 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 Routine Airborne Releases over Time

We evaluated the releases of radioactivity over time to identify those years during which routine airborne releases were highest. At the ICPP, gaseous and particulate radioactive material discharged from the main stack were sampled at the 90-ft (27-m) 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 1979, the flow measurement instrumentation and the sampling system for gaseous radionuclides were updated. 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 used two-day cooled fuel, was much more hazardous and required special handling. For later years, Wong and Roberts (1979) provided 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 , ^{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) stack, 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 ft 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 ft 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). Releases of liquid wastes in the early years were discharged 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 (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 ft 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 Liquid Effluents.xls). Historic documents show that 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 referenced 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.

Liquid Wastes from the ICPP

The ICPP was primarily designed to recover enriched uranium from spent reactor fuel elements. 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. Large volumes of other liquids, containing low concentrations of radioactive materials, were diluted and discharged to a 592-ft 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. The 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 pipe inside a carbon steel shell and penetrated about 140 ft below the water table (Dickey et al. 1972).

Quarterly effluent monitoring reports summarized the liquid radioactive releases from the ICPP 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).

Tritium contributed over 99% of the total radioactivity in the ICPP 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 New Waste Calcining Facility (NWCF) and Fluorinel Dissolution Process and Fuel Storage Facility (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). In April 1983, tritium releases increased due to operation of the liquid

waste evaporator and the APS condenser (ENICO 1983b). Iodine-129 release trends were similar to those for tritium in the 1980s (ENICO 1983b).

Test Reactor Area Liquid Wastes

TRA is a complex with three test reactors, MTR, ETR, and ATR, that used about 150 million gallons of water per month for reactor cooling, irrigation, and domestic use supplied from deep wells into the Snake River Plain Aquifer (about 450 ft 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 through 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 Stationary Low-Power Reactor No. 1 (SL-1) Burial Ground (one trench and two pits 1600 ft east of old 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 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 ^{233}U activity above 10 nanocuries per gram were stored aboveground in 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 1958, 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 ft of waste with a reported activity of over

35,000 Ci from the Rocky Flats Plant in Colorado, was buried at the INEEL RWMC. 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 methods of disposal. For trench disposal 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, and were about 50 to 300 ft wide, 250 to 100 ft long and 5 to 15 ft 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 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 (30 ft wide by 350 ft long by 5 ft high) was embedded in compacted soil, and extended 4 inches above a thick asphalt pad. Figures 11 and 12 compare the types of wastes located at the RWMC and the source of the solid wastes at the INEEL (Osloond 1970; Smith 1981). The Naval Reactor Facility and Test Reactor Area contributed large volumes of solid waste; waste at the ANL-W underground facility had the highest activity, based on data from Osloond (1970).

Several major events at the INEEL and specifically at the RWMC led to modifications in the procedures and burial practices at the Site. After the SL-1 accident in January 1961, the SL-1 Burial Ground, about ¼ mile from the reactor location, was opened to receive waste from that accident. However, some of the waste was put into Pit 1 at the RWMC, which was reopened in October 1961 to accept this SL-1 accident waste. In February 1962, the RWMC suffered a severe flood when 2 inches of rain and 8 inches of snow fell in three days. When a warming trend followed this snow and with the upper foot of the ground frozen, extensive runoff occurred into open pits and trenches that contained boxes and barrels of radioactive waste. Pits 2 and 3 and Trenches 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.

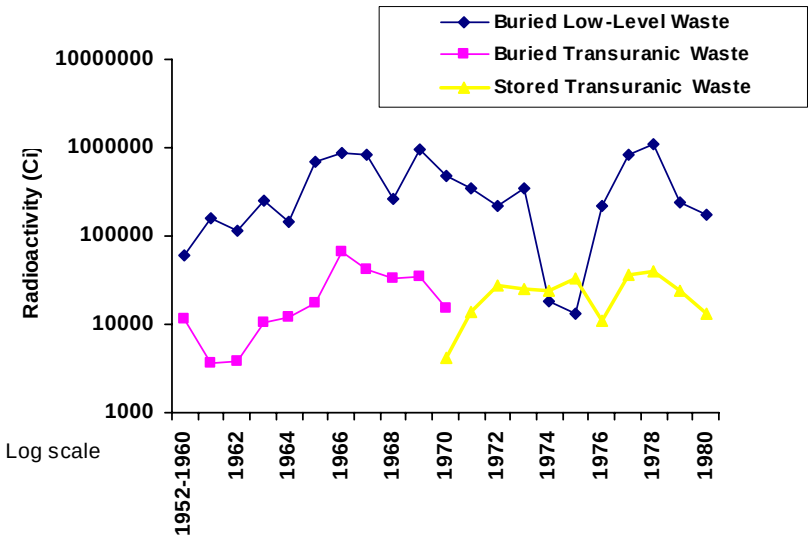


Figure 11. Categories of solid wastes at the Radioactive Waste Management Complex at the INEEL. In the early 1970s, transuranic wastes were stored in above ground facilities and were no longer buried.

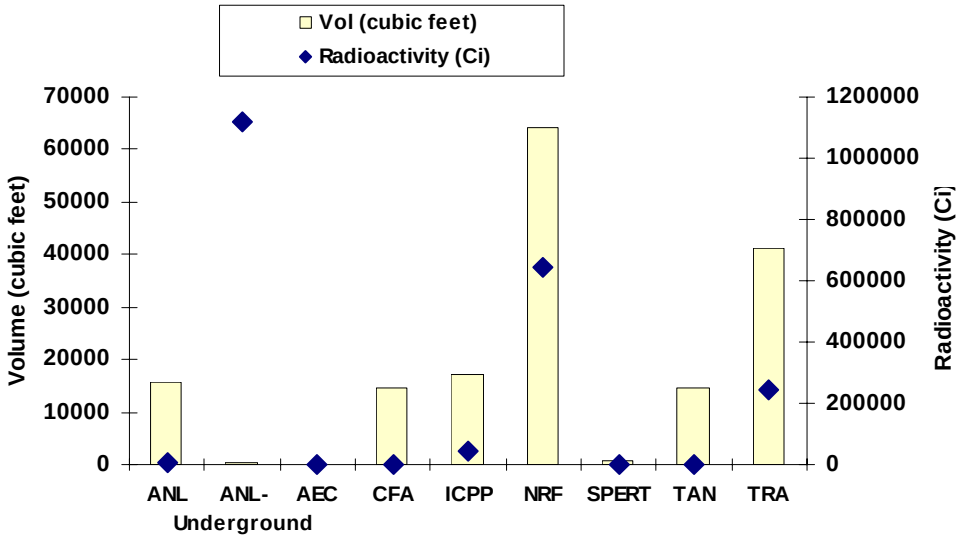


Figure 12. Sources of solid waste at the INEEL in 1969. The columns represent the volume of waste (left axis) and the filled diamond shapes represent the level of radioactivity in the waste (right axis).

The 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 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. The monthly volume of solid radioactive waste varied between 50 and 200 m^3. Examples of the radioactivity levels in the waste 20,000 Ci in December 1981, about 50,000 Ci in May, June, and July 1982, and 1000 Ci in October 1983. 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 at the INEEL (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, which we evaluated as a potential exposure pathways in the Groundwater Pathway section of this report.

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 provided 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-ft 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 ft 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 ft 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 ft 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 Sondrup 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 Sondrup 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 measured by the USGS, and the data for each radionuclide are referenced within each section below.

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 likely 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[5] 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.

[5] 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-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 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 were easier to detect.

To support 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 measured in the 1981 plume than in the 1977 plume. This observation seemed to support the hypothesis of 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 concluded 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 use 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 ft 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 13) 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, tritium was not detected offsite in groundwater. Since that time, it appears that the tritium plume has receded, and it is speculated that radioactive decay (tritium $T_{1/2}$ = 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.

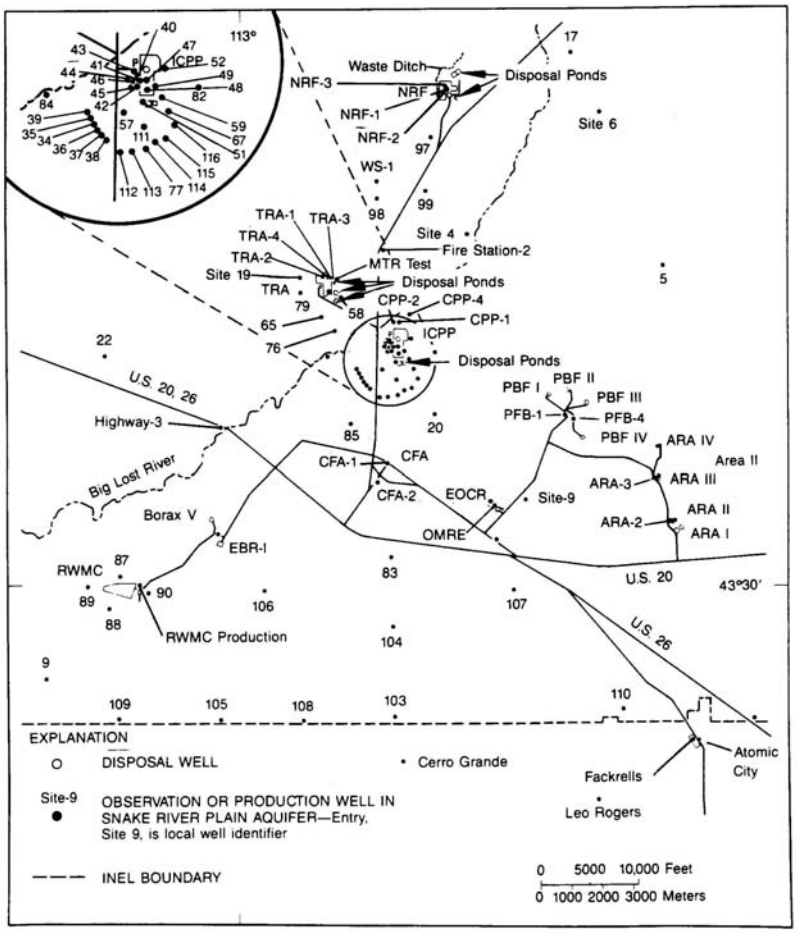


Figure 13. Snake River Plain Aquifer observation wells at the INEEL. Wells 103, 105, and 108, closest to the INEEL southern boundary (dotted line), are the wells that contained measurable quantities of tritium in 1983-1985.

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.

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[6]. 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.

$$1.2 \text{ pCi mL}^{-1} \cdot 37000 \frac{\text{Bq m}^{-3}}{\text{pCi mL}^{-1}} = 44000 \text{ Bq m}^{-3} \cdot 1.4\times10^{-11} \text{ Sv per Bq m}^{-3}$$

$$= 6.2\times10^{-7} \text{ Sv} = 0.06 \text{ mrem}$$

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. The dose calculated here is significantly less than the annual dose limit of 10 mrem.

[6] 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.

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}.

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

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

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

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.

Ranking Method 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 ranking process helps to focus the research efforts so that resources will be allocated to the radionuclides, time periods, and events that contribute most to doses to people 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 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 exposure pathway so the primary 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 ranking 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 NCRP 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	2×10^{-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 (see the Episodic Release Evaluation section for more details on this model). This model assumed a flat terrain, similar to the INEEL area. If we assumed a air release from a 75 m stack (comparable to the ICPP) as a conservative approach, then the atmospheric concentration, *C*, of a particular radionuclides was calculated as follows:

$$C = \frac{f\,Q}{\pi\, u\, \sigma_y\, \sigma_z} \exp[-1/2(H/\sigma_z)^2] \qquad (1)$$

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})
σ_y = horizontal dispersion coefficient at coordinate x (m)
σ_z = vertical dispersion coefficient at coordinate (m)
H = height of effluent release (m).

For ranking 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):

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

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

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 (σ_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 (σ_y) was 924 m and the vertical dispersion coefficient (σ_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. For f, we assumed that the wind blows 100% of the time toward the potentially exposed individual, so f =1.

The INEEL has 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 at 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 HDE (Historic Dose Evaluation (DOE 1991a) provided dose calculations 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. For the Task Order 5 ranking process, 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 was 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. They also reviewed 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 for their impact on the calculated doses and treated each radionuclide independently (DOE 1991a, Appendix A)
- The Site recalculated releases of the noble fission gases, krypton and xenon, from the TRA from 1952 through 1968 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 RSAC 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 had reported just the opposite: the release of ^{137}Xe greater than the release of ^{138}Xe.
    - To correct this anomaly, the Site revised the original estimates (for the xenon and krypton gases from TRA). They based the revised estimates on the reported total annual airborne effluent activity compiled from 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, they used the proportions applicable to ATR based on the isotopic composition for 1987 airborne effluent (DOE 1991a).
- 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 tabulated reported annual data 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), and for total releases of specific radionuclides from INEEL for assessing the potential exposures at offsite and onsite locations (see OffsiteAirRanking.xls and OnsiteAirRanking.xls). As we compiled release data for specific radionuclides, we noted the source of the estimates or the special calculations used to provide a release estimate. Table 4 summarizes the source of the release estimates for all radionuclides for the years of consideration.

Table 4. Summary of Air Source Term Information

Radionuclide	Half-life	Documentation of source term methods
^{3}H	12.35 y	1952-1989: DOE/ID-12119 Vol. 2, 1991. 1990-1992: INEL Site Environmental Reports for 1990-1992.
^{14}C	5730 y	1978-1989: DOE/ID-12119 Vol. 2, 1991. 1990-1992: INEL Site Environmental Reports for 1990-1992. 1952-1977: used lowest $^{41}Ar/^{14}C$ ratio of reported releases from 1978-1992, combined with ^{41}Ar reported releases to estimate ^{14}C releases.
^{13}N	9.97 min	1972: DOE/ID-12119 Vol. 2, 1991. 1974: ERDA-1536, 1977 1952-1971, 1973, 1975-1992: assumed releases equal to twice the average of reported releases for other years.
^{24}Na	15 h	1952-1968, 1970, 1989: DOE/ID-12119 Vol. 2, 1991. 1969, 1971-1985: assumed releases equal to average releases for other years. 1990-1992: INEL Site Environmental Reports for 1990-1992.
^{41}Ar	1.8 h	1952-1989: DOE/ID-12119 Vol. 2, 1991. 1990-1992: INEL Site Environmental Reports for 1990-1992.
^{51}Cr	27.8 d	1952-1969, 1985-1989: DOE/ID-12119 Vol. 2, 1991. For 1970-1984: assumed releases equal to twice the arithmetic mean (0.11 curies) of reported releases for other years. 1990-1992: INEL Site Environmental Reports for 1990-1992.
^{54}Mn	303 d	1965, 1973-1975: DOE/ID-12119 Vol. 2, 1991. For other years: assumed releases equal to twice the arithmetic mean of reported releases.
^{60}Co	5.27 y	1952-1968, 1971-1974, and 1987: DOE/ID-12119 Vol. 2, 1991. 1969, 1970, 1975-1986: assumed releases equal to the average of reported annual releases after 1968. 1988-1992: RWMIS reports
^{76}As	26.5 hr	1975: DOE/ID-12119 Vol. 2, 1991. 1952-1974, 1976-1992: assumed releases as twice that reported in 1975.
^{82}Br	35.3 hr	1990-1992: assumed the average annual release for 1987-1989 1973–1989: DOE/ID-12119 Vol. 2, 1991. 1961–1972: assumed releases as twice the annual releases for 1973–1989 1952-1960: no releases because EBR-II at ANL-W (source of ^{82}Br) came online in 1961.
^{85}Kr	10.7 y	1952-1989: DOE/ID-12119 Vol. 2, 1991. 1990-1992: INEL Site Environmental Reports for 1990-1992.
^{85m}Kr	4.48 hr	1952-1968, 1974-1989: DOE/ID-12119 Vol. 2, 1991. 1990-1992: INEL Site Environmental Reports for 1990-1992.

Table 4. Summary of Air Source Term Information (continued)		
^{87}Kr	1.27 hr	1952-1968, 1971-1989: DOE/ID-12119 Vol. 2, 1991. 1990-1992: INEL Site Environmental Reports for 1990-1992.
^{88}Kr	2.86 hr	1952-1989: DOE/ID-12119 Vol. 2, 1991. 1990-1992: INEL Site Environmental Reports for 1990-1992.
^{89}Kr	3.2 m	1969-1974, 1978: DOE/ID-12119 Vol. 2, 1991. 1952-1968, 1975-1977, 1979-1992: assumed releases equal to twice the annual average release for 1969-1974, and 1978.
^{88}Rb	17.8 min	1952-1968: DOE/ID-12119 Vol. 2, 1991 (Tables A-7 through A-10) 1969-1973, 1975-1977: assumed release relationship between Kr-88 and Rb-88 in DOE/ID-12119 Vol. 2, 1991 to estimate releases from 1952-1968. 1974: ERDA-1536, 1977. 1978-1990: INEL Site Environmental Reports for 1990-1992.
^{89}Rb	15.4 min	1952-1974, 1986-1989: DOE/ID-12119 Vol. 2, 1991. 1975-1985: assumed releases equal to average of reported releases from 1970-1992 1990-1992: INEL Site Environmental Reports for 1990-1992.
^{89}Sr	52 d	1953-1964: DOE/ID-12119 Vol. 2, 1991. 1965-1992: assumed ^{89}Sr releases were combined and reported with the ^{90}Sr releases.
^{90}Sr	27.7 y	1952-1989: DOE/ID-12119 Vol. 2, 1991. 1990-1992: INEL Site Environmental Reports for 1990-1992.
^{91}Sr	9.7 h	1957-1963, 1979: DOE/ID-12119 Vol. 2, 1991. 1952-1955, 1964-1978, 1980-1992: assumed releases equal to total airborne annual release times average ratio of ^{91}Sr to total airborne releases for years when ^{91}Sr releases were reported. Average ratio (^{91}Sr/total releases) is 3.6 E-07.
^{91}Y	58.8 d	1953-1963: DOE/ID-12119 Vol. 2, 1991. 1964-1992: assumed releases equal to total airborne annual release times average ratio of ^{91}Y to total airborne releases for years when ^{91}Y releases were reported.
^{91m}Y	50.3 m	1952-1968, 1988-1989: DOE/ID-12119 Vol. 2, 1991. 1969-1987: assumed releases equal to total airborne annual release times average ratio of ^{91m}Y to total airborne releases for years when ^{91m}Y releases were reported. 1990-1992: INEL Site Environmental Reports for 1990-1992.
^{95}Zr	65.5 d	1953-1976: DOE/ID-12119 Vol. 2, 1991. 1977-1992: assumed releases equal to average reported releases from 1970-1977.
^{99}Tc	213000 y	1952-1989: No measurements of ^{99}Tc were made at the INEEL; estimated atmospheric release using fission product ratios (Till 1984) of ^{90}Sr/^{99}Tc of 5300 and ^{137}Cs/^{99}Tc of 7600. These ratios were applied to measured releases of ^{90}Sr and ^{137}Cs to obtain the ^{99}Tc release estimates with the largest release estimate calculated with these ratios used.
^{99m} Tc	6.01 h	1952-1968, 1988: DOE/ID-12119 Vol. 2, 1991. 1969-1987: assumed release equal total airborne annual release times average ratio of ^{99m}Tc to total airborne releases for years when ^{99m}Tc released were reported. 1989: Average of 1988, 1990-1992 1990-1992: INEL Site Environmental Reports for 1990-1992.
^{103}Ru	39.4 d	1953-1963, 1971-1973: DOE/ID-12119 Vol. 2, 1991. 1964-1970, 1974-1992: Assumed release equal to total airborne annual release times average ratio of ^{103}Ru to total airborne releases for years when ^{103}Ru releases were reported.
^{106}Ru	1 y	1952: Assumed releases same as 1953. 1953-1982, 1984, 1987-1989: DOE/ID-12119 Vol. 2, 1991. 1983, 1985-1986: assumed releases equal to average of reported releases for 1980-1992. 1990-1992: INEL Site Environmental Reports for 1990-1992.

Table 4. Summary of Air Source Term Information (continued)		
^{125}Sb	2.7 y	1965, 1966, 1968-1989: DOE/ID-12119 Vol. 2, 1991. 1952-1964, 1967: assumed releases equal to twice the average for 1968-1992. 1990-1992: INEL Site Environmental Reports for 1990-1992.
^{132}Te	3.26 d	1952-1955, 1964-1966, 1968-1992: Assumed annual release of ^{132}Te equal to total airborne annual release times average ratio of ^{132}Te to total airborne releases for years when ^{132}Te releases were reported. Average ratio (^{132}Te /total releases) is 1.1E-05. 1956-1963, 1967: DOE/ID-12119 Vol. 2, 1991.
^{129}I	15700000 y	1952-1955: assumed releases equal to twice the average for 1956-1960. 1956-1989: DOE/ID-12119 Vol. 2, 1991. 1990-1992: INEL Site Environmental Reports for 1990-1992.
^{131}I	8.04 d	1952-1989: DOE/ID-12119 Vol. 2, 1991. 1983: Assumed average of reported releases from 1981-1985. 1990-1992: INEL Site Environmental Reports for 1990-1992.
^{132}I	2.28 h	1952-1955, 1964, 1965, 1968-1977, 1979-1984: Assumed annual release of ^{132}I equal to total airborne annual release times average ratio of ^{132}I to total airborne releases for years when ^{132}I releases were reported. Average ratio (^{132}I /total releases) is 5.8E-04. 1956-1963, 1966-1967, 1978, 1985, 1987-1989: DOE/ID-12119 Vol. 2, 1991. 1986, 1990-1992: INEL Site Environmental Reports for 1990-1992. 1987: Assumed average of 1985-1992.
^{133}I	20.9 h	1952-1968, 1976, 1978, 1985, 1988,1989: DOE/ID-12119 Vol. 2, 1991. 1969-1975, 1977, 1979: Assumed annual release of ^{133}I equal to total airborne annual release times average ratio of ^{133}I to total airborne releases for years after 1963 when ^{133}I releases were reported. Average ratio for that period (^{133}I /total releases) is 1.1E-06. 1986, 1987, 1990-1992: INEL Site Environmental Reports for 1990-1992.
^{134}I	52.5 min	1978: DOE/ID-12119 Vol. 2, 1991. 1952-1977, 1979, 1989, 1991, 1992: Assumed annual releases equal to twice the release reported in 1978. 1990: INEL Site Environmental Reports for 1990.
^{135}I	6.61 h	1952-1955, 1964, 1965, 1968-1977, 1979-1984, 1986-1991: Assumed annual release of ^{135}I equal to total airborne annual release times average ratio of ^{135}I to total airborne releases for years when ^{135}I I releases were reported. 1956-1963, 1966, 1967, 1978, 1985: DOE/ID-12119 Vol. 2, 1991. 1992: INEL Site Environmental Reports for 1992.
^{131m}Xe	12 d	1952-1955, 1964-1984, 1986-1989, 1991, 1992: Assumed annual release of ^{131m}Xe equal to total airborne annual release times average ratio of ^{131m} Xe to total airborne releases for years when ^{131m} Xe releases were reported. 1956-1963, 1985, 1990: DOE/ID-12119 Vol. 2, 1991
^{133}Xe	5.25 d	1952-1969, 1971: DOE/ID-12119 Vol. 2, 1991 1970: Assumed average release for 1968-1973. 1990-1992: INEL Site Environmental Reports for 1990-1992.
^{133m} Xe	2.26 d	1952-1955, 1964-1976, 1980-1984, 1986-1989, 1991, 1992: Assumed annual release of ^{133m}Xe equal to total airborne annual release times average ratio of ^{133m}Xe to total airborne releases for years when ^{133m}Xe releases were reported. 1956-1963, 1977-1979, 1984, 1985, 1990: DOE/ID-12119 Vol. 2, 1991
^{135}Xe	9.1 h	1952-1989: DOE/ID-12119 Vol. 2, 1991 1969, 1970: Assumed releases equal to average of releases reported for 1967-1972. 1990-1992: INEL Site Environmental Reports for 1990-1992.

Table 4. Summary of Air Source Term Information (continued)		
^{135m}Xe	15.6 min	1952-1968, 1974-1989: DOE/ID-12119 Vol. 2, 1991. 1969-1973: Assumed releases equal to average of releases reported for 1966-1977. 1990-1992: INEL Site Environmental Reports for 1990-1992.
^{138}Xe	17.5 min	1952-1989: DOE/ID-12119 Vol. 2, 1991. 1990-1992: INEL Site Environmental Reports for 1990-1992.
^{134}Cs	2.06 y	1952-1979, 1981: DOE/ID-12119 Vol. 2, 1991. 1980, 1982-1992: Assumed releases equal to twice the average of releases for 1977-1992.
^{136}Cs	13 d	1956-1963 (RaLa Runs): DOE/ID-12119 Vol. 2, 1991.
^{137}Cs	30.1 y	1952-1989: DOE/ID-12119 Vol. 2, 1991. 1990-1992: INEL Site Environmental Reports for 1990-1992.
^{138}Cs	32.2 min	1952-1989: DOE/ID-12119 Vol. 2, 1991. 1990-1992: INEL Site Environmental Reports for 1990-1992.
^{139}Ba	83 m	1952-1968, 1974-1989: DOE/ID-12119 Vol. 2, 1991. 1969-1973: Assumed release equal to average of releases reported for 1965-1977. 1990-1992: INEL Site Environmental Reports for 1990-1992.
^{140}Ba	12.8 d	1952-1968, 1979: DOE/ID-12119 Vol. 2, 1991. 1969-1978: Assumed release equal to average of releases reported for 1964-1968. 1980-1989: Assumed release equal to average of releases reported for 1979-1992. 1990, 1992: INEL Site Environmental Reports for 1990, 1992.
^{141}Ce	32.5 d	1952-1968, 1972, 1975, 1976: DOE/ID-12119 Vol. 2, 1991. 1969-1971, 1973, 1974, 1977-1992: Assumed annual release of ^{141}Ce equal to total airborne annual release times average ratio of ^{141}Ce to total airborne releases for years after 1963 when ^{141}Ce releases were reported. Average ratio for that period (^{141}Ce /total releases) is 6.9E-07.
^{144}Ce	284 d	1952: Assumed release same as 1953 reported release. 1953-1979, 1982: DOE/ID-12119 Vol. 2, 1991. 1980, 1981, 1983-1992: Assumed release equal to average of releases reported for 1976-1982.
^{143}Pr	13.6 d	1956-1963 (RaLa Runs): DOE/ID-12119 Vol. 2, 1991.
^{147}Pm	2.62 y	1953-1963 (RaLa Runs): DOE/ID-12119 Vol. 2, 1991.
^{154}Eu-	16 y	1953, 1956-1959, 1962, 1963, 1972-1975, 1977: DOE/ID-12119 Vol. 2, 1991.
^{203}Hg	47 d	1952-1968: DOE/ID-12119 Vol. 2, 1991. 1968-1986: Assumed annual release of ^{203}Hg equal to total airborne annual release times average ratio of ^{203}Hg to total airborne releases for years when ^{203}Hg releases were reported. Average ratio (^{203}Hg /total releases) is 1.1E-05. 1987: INEL Site Environmental Report for 1987. 1988-1992: Assumed same release as in 1987.
^{238}U + D	4.5E9 y	1952-1987, 1990-2992: Assumed release equal to a hundred times releases reported in 1988. 1988, 1989: DOE/ID-12119 Vol. 2, 1991.
^{238}Pu	86.4 y	1953-1989: DOE/ID-12119 Vol. 2, 1991. 1990-1992: INEL Site Environmental Reports for 1990-1992, reported as total Pu.
^{239,240}Pu	24390 y	1953-1989: DOE/ID-12119 Vol. 2, 1991. 1990-1992: INEL Site Environmental Reports for 1990-1992, reported as total Pu.

Where data were not reported or where a specific radionuclide had not been monitored, 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 ranking calculations. Table 4 and the Excel spreadsheets that contain these data note the source of the annual release estimates for each radionuclide, whether reported directly in a site document, or whether other methods were utilized (see OffsiteAirRanking.xls and OnsiteAirRanking.xls). We provide some examples in the next paragraphs

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 420 to 26,000 with a geometric mean of 3200. To ensure a conservative approach to estimating ^{14}C releases we used the lowest ratio (420), which would maximize the ^{14}C release estimate, and divided the measured annual releases of ^{41}Ar by this ratio. Table 5a shows the calculated $^{41}Ar/^{14}C$ ratios, and Table 5b shows the calculated ^{14}C releases based on the average ratio of $^{41}Ar/^{14}C$.

Table 5a. Calculated $^{41}Ar/^{14}C$ Ratios for Years When Both Were Measured [a]

	Measured releases (Ci)[b]		
Year	^{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] To conservatively estimate ^{14}C releases for years before 1978 when ^{14}C releases were not reported, 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.

b From DOE (1991a).

Table 5b. Calculated ^{14}C Releases Based on Reported ^{41}Ar Releases

	Releases to air (Ci)	
	Reported[b]	Calculated[a]
Year	^{41}Ar	^{14}C
1952	75300	179
1953	166000	395
1954	122000	290
1955	181000	431
1956	88900	212
1957	34200	81
1958	46500	111
1959	75800	180
1960	104000	248
1961	150000	357
1962	157000	374
1963	269000	640
1964	255000	607
1965	234000	557
1966	150000	357
1967	89100	212
1968	56800	135
1969	55000	131
1970	28000	67
1971	16000	38
1972	7200	17
1973	4500	11
1974	4100	10
1975	5100	12
1976	5000	12
1977	3400	8

[a] To conservatively estimate ^{14}C releases for years before 1978 when ^{14}C releases were not reported, 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.

b From DOE (1991a).

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 applied 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 6 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 6. 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+01	6.7E-03	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+01	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	5.67	7.5E-04	2.8E-03
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. For all years see TO5-SpecialCalculations.xls.

[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.

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 ^{129}I 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; the organic form is less important for dose to local residents because it does not deposit as readily on vegetation. For our ranking

methods, we assumed all ^{129}I was in the elemental form and used the NCRP 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 as twice the arithmetic mean of measured releases from 1965–1992. 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 from 1952–1981. For 1982–1992, we assumed annual releases equal to twice the annual average ^{134}Cs releases for 1952–1981.

Releases of the activation product, ^{99m}Tc, were reported from the INEEL for 1952-1968 and 1988-1992. For other years we assumed releases of ^{99m}Tc equal to total airborne annual release times the average ratio of reported releases of ^{99m}Tc to total airborne releases for years when ^{99m}Tc releases were reported. The average ratio of ^{99m}Tc to total airborne releases for years when both were reported is 6.8 E-06. Releases of the activation product, ^{60}Co, were reported from 1952-1968, 1971-1974, and in 1987. For ranking purposes, we assumed annual releases from 1975 through 1986, and 1988-1992 as the arithmetic mean of reported releases. For special calculations of release estimates see FacilityAirRelease.xls.

Results of Screening Routine Releases

Radionuclides released to air from the INEEL facilities were conservatively[7] assessed in several ways to determine the radionuclides that ranked highest onsite and offsite exposures for all years of operations and for individual years. The ranking process also identified the individual years that were most important for onsite and offsite exposures, the radionuclides that ranked highest for individual years, and the facilities that ranked highest in terms of potential exposure.

All results using the NCRP methods are reported. In addition, ranking values using the RSAC program are presented for selected years in support of the results from the NCRP methodology. 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 ranking calculation was a list of values for individual radionuclides for each year that provided the basis for ranking the radionuclides. We summed the ranking values and calculated the relative ranking value for each radionuclide and a total relative ranking value

[7] 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.

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 and the RSAC program were compiled in Excel spreadsheets for each screening scenario (see OffsiteAirRanking.xls and OnsiteAirRanking.xls). Based on these results, we ranked the radionuclides, the years, and the facilities that contributed significantly to the screening values for routine releases.

Ranking the Relative Importance of Radionuclides

Figures 14 and 15 show the results of the offsite and onsite ranking of all radionuclides from 1952–1992, respectively. All pathways (inhalation; plume immersion; ground contamination; and ingestion of meat, vegetables, and milk from areas exposed to contaminants) were considered for the offsite location. Only the inhalation and plume immersion pathways were considered for the onsite location. When considering all radionuclides released from the INEEL for all years, this figure shows that ^{137}Cs, ^{131}I and ^{90}Sr ranked highest at the offsite location. The agreement between the NCRP and RSAC results confirm that the NCRP methodology is a valid method for identifying the radionuclides and years that were most important in terms of public health.

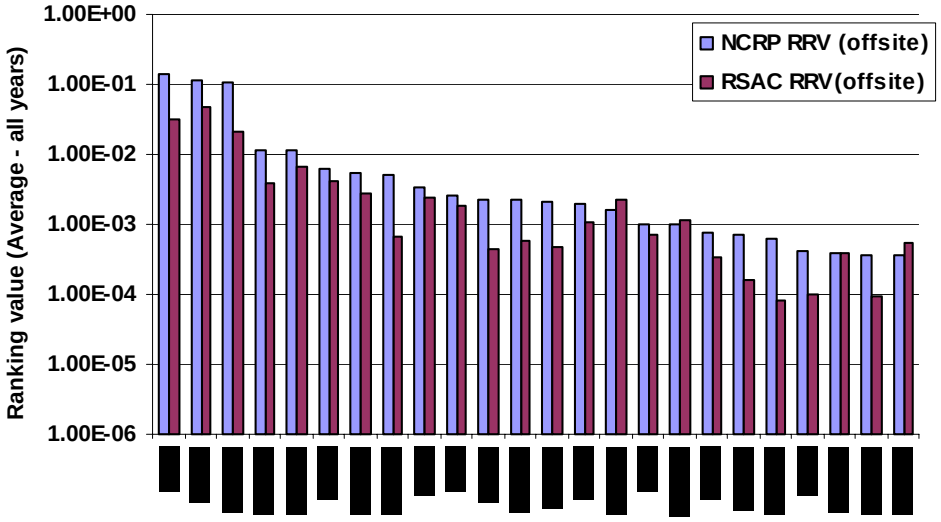


Figure 14. Results of ranking radionuclides released from routine operations at the INEEL for 1952–1992 considering all pathways of exposure in the NCRP methodology at the *offsite* location. For the radionuclides ranked highest, the RSAC code confirmed the general ranking obtained using the NCRP methods.

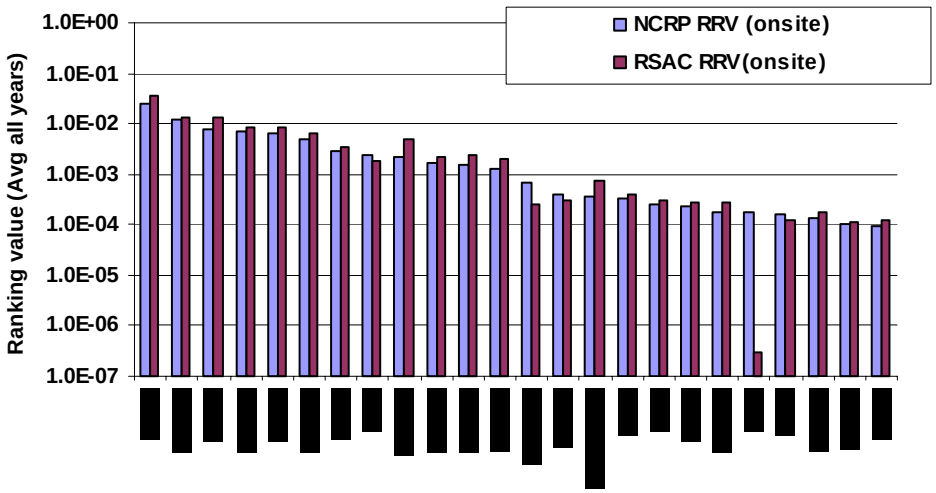


Figure 15. Results of ranking radionuclides released from routine operations at the INEEL for 1952–1992 considering the inhalation and plume immersion pathways at the *onsite* location. For the radionuclides ranked highest, the RSAC code confirmed the general ranking obtained using the NCRP methods.

Similar agreement was noted between the NCRP and RSAC methods for ranking the relative importance of radionuclides at the onsite location (Figure 15). For onsite exposures where plume immersion and inhalation were the essential pathways, releases of ^{41}Ar, ^{138}Xe, and ^{88}Kr ranked highest when considering all years. We see a similar pattern of agreement for individual years,, as well. For example, Figures 16 and 17 shows ranking results at offsite and onsite locations for 1957, respectively, where we observe close agreement between the two methods. For individual years, we see that the highest ranked radionuclides can vary somewhat depending on the particular program or events occurring that year. In 1957, when the Radioactive Lanthanum (RaLa) program began, the highest ranked radionuclides were ^{131}I, ^{90}Sr, ^{41}Ar , and ^{144}Ce at the onsite location (Figure 17). At the offsite location, the same radionuclides ranked high (^{137}Cs, ^{131}I and ^{90}Sr) in 1957 as for all years evaluated together (Figure 14).

Although there is close agreement between the results obtained with the NCRP and RSAC methods, Figures 14, 15, 16, and 17 also show that there are individual differences in the ranking order for specific radionuclides that are ranked lower. Some of these differences are related to differences in parameter values and in how the programs handle decay products. Ranking results for the top ranked radionuclides for all years of operations are shown in Tables 7 and 8, respectively. For the offsite location, the radionuclides, ^{131}I, ^{90}Sr, ^{137}Cs, and ^{106}Ru ranked the highest with respect to potential for exposure when all pathways were considered. For the onsite location, ^{41}Ar, ^{138}Xe, and ^{88}Kr ranked highest, but ^{144}Ce, ^{90}Sr and ^{106}Ru were also among the radionuclides that ranked high. The ratio of NCRP to RSAC ranking values shows the general agreement between the methods. A ratio of one would indicate perfect agreement between the two methods. In Table 7, the ratios indicate that the NCRP screening method tends to provide higher ranking estimates than the RSAC code when all pathways of exposure are considered. When only the inhalation and plume immersion pathways are considered at the onsite location, Table 8 shows that the NCRP/RSAC ratios of ranking values tend to be less than one. Despite

these differences in agreement in a few cases, however, the unmistakable message is that the NCRP screening method was a valid method for identifying the radionuclides and years that were most important in terms of public health.

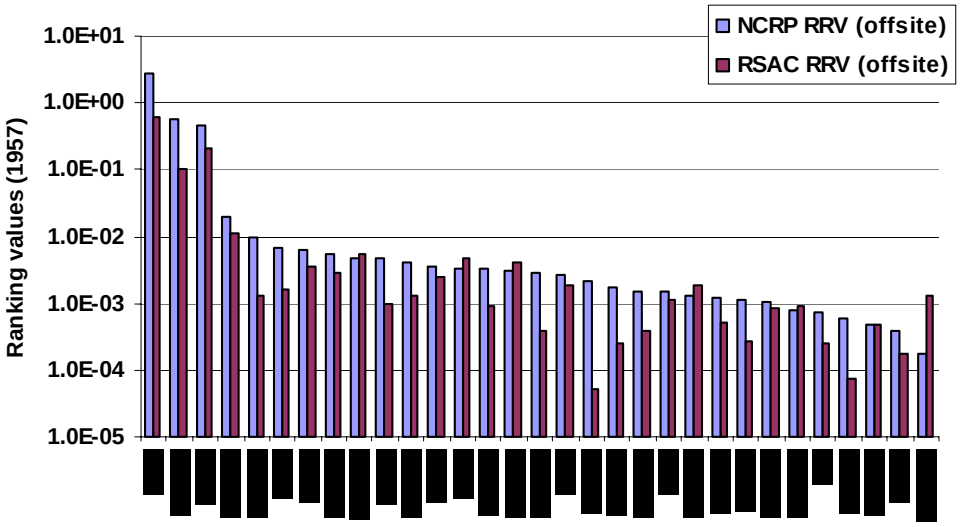


Figure 16. Results of ranking radionuclides released from operations at the INEEL during 1957, considering all exposure pathways at the *offsite* location. In general, the ranking results from the RSAC method confirmed the relative ranking obtained using the NCRP methods.

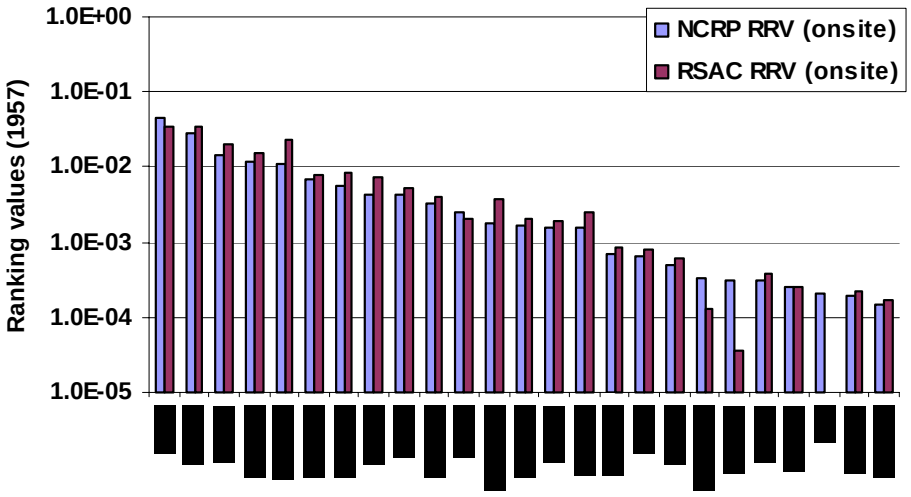


Figure 17. Results of ranking radionuclides released from operations at the INEEL during 1957, considering the inhalation and plume immersion pathways at the *onsite* location. In general, the ranking results from the RSAC method confirmed the relative ranking obtained using the NCRP methods.

Table 7. Ranking Values for Exposure to Radionuclide Releases for 1952-1992 for All Pathways at the Offsite Location [a]

Radionuclide	Half-life	Total releases 1952-1992 (Ci)	Relative ranking value NCRP	RSAC	Ratio[b] NCRP/RSAC
^{131}I	8.04 d	2850	0.14	0.032	4.2
^{90}Sr	27.7 y	340	0.11	0.049	2.3
^{137}Cs	30.1 y	292	0.11	0.022	5.1
^{106}Ru	1 y	690	0.012	0.038	3.1
^{144}Ce	284 d	1200	0.0011	0.0065	1.8
^{41}Ar	1.8 h	2,420,000	0.0062	0.0042	1.5
^{134}Cs	2.06 y	25	0.0055	0.0029	1.9
^{138}Xe	17.5 min	1,260,000	0.0049	0.00068	7.2
^{129}I	15700000 y	3.5	0.0033	0.0024	1.4
^{14}C	5730 y	5700	0.0025	0.0019	1.4
^{89}Sr	52 d	105	0.0022	0.00046	4.8
^{125}Sb	2.7 y	79.6	0.0023	0.00058	3.7
^{60}Co	5.27 y	7.1	0.0021	0.00046	4.4
^{95}Zr	65.5 d	265	0.0019	0.0011	1.8

[a] Of over 50 radionuclides evaluated, these radionuclides ranked highest at the offsite location, when all pathways of exposure were considered.

[b] A ratio of one would indicate perfect agreement between the two methods. These ratios indicate that the NCRP screening method tends to provide higher ranking estimates than the RSAC code when all pathways of exposure are considered.

Table 8. Ranking Values for Exposure to Radionuclide Releases for1952-1992 from the Inhalation and Plume Immersion Pathways at the *Onsite* Location [a]

Radionuclide	Half-life	Total releases (1952-1992)Ci	Relative ranking value NCRP	RSAC	Ratio NCRP/RSAC
Ar-41	1.8 h	2.4E+06	0.024	0.035	0.7
Xe-138	17.5 min	1.3E+06	0.012	0.014	0.9
Kr-88	2.86 hr	4.0E+05	0.008	0.014	0.6
Ce-144	284 d	1.3E+03	0.007	0.009	0.8
Sr-90	27.7 y	3.4E+02	0.007	0.008	0.8
Ru-106	1 y	7.0E+02	0.005	0.006	0.8
Kr-87	1.27 hr	4.1E+05	0.003	0.004	0.8
I-131	8.04 d	2.9E+03	0.002	0.002	1.3
Pu-238	86.4 y	6.4E-01	0.0022	0.0047	0.5
Pm-147	2.62 y	8.1E+02	0.0017	0.0022	0.8
Xe-133	5.25 d	5.3E+06	0.0016	0.0024	0.6
Xe-135	9.1 h	6.1E+05	0.0012	0.0021	0.6
Xe-135m	15.6 min	2.7E+05	0.0007	0.0003	2.6
Rb-89	15.4 min	3.0E+04	0.0004	0.0003	1.3
Pu239,240		9.5E-02	0.00036	0.00076	0.5
I-132	2.28 h	1.2E+04	0.00032	0.00040	0.8
Y-91	58.8 d	3.3E+02	0.00025	0.00030	0.8
Sr-89	52 d	1.0E+02	0.00023	0.00028	0.8
Kr-85m	4.48 hr	1.1E+05	0.00017	0.00028	0.6

[a] Of over 50 radionuclides evaluated, these radionuclides ranked highest at the offsite location when the inhalation and plume immersion pathways were considered.

[b] A ratio of one would indicate perfect agreement between the two methods. These ratios indicate that the NCRP screen method tends to provide lower ranking estimates than the RSAC code when these pathways of exposure are considered

Ranking the Relative Importance of Years

Figures 18 and 19 show the results of the relative ranking of the releases by years for both offsite and onsite locations. For offsite exposures, the ranking results showed the years, 1957, 1958, and 1959, had the highest screening values (Figure 18). At the onsite location, 1963, 1964, and 1965 had the largest screening values (Figure 19). 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.

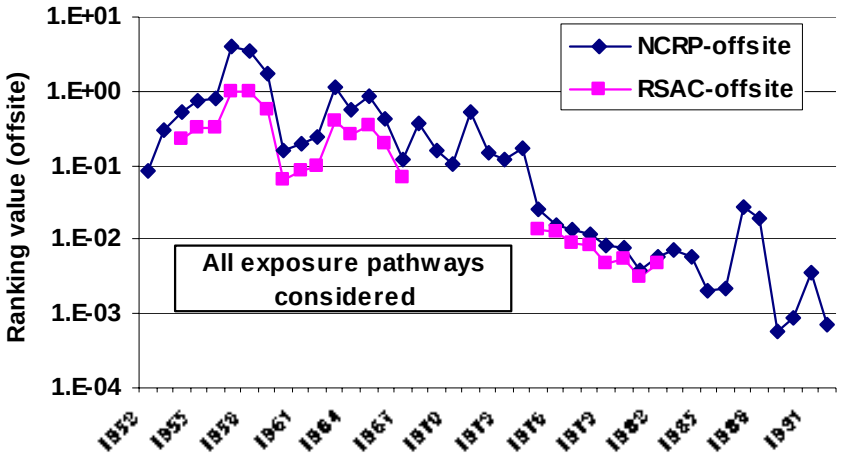


Figure 18. Results of ranking individual years for all radionuclides for all exposure pathways from 1952–1992. The ranking value provides a relative way to evaluate the years that had the largest offsite exposure. The years 1957, 1958, and 1959 had the highest ranking values at the offsite location. The RSAC code was used to estimate ranking values for several years to confirm the relative ranking values obtained with the NCRP method.

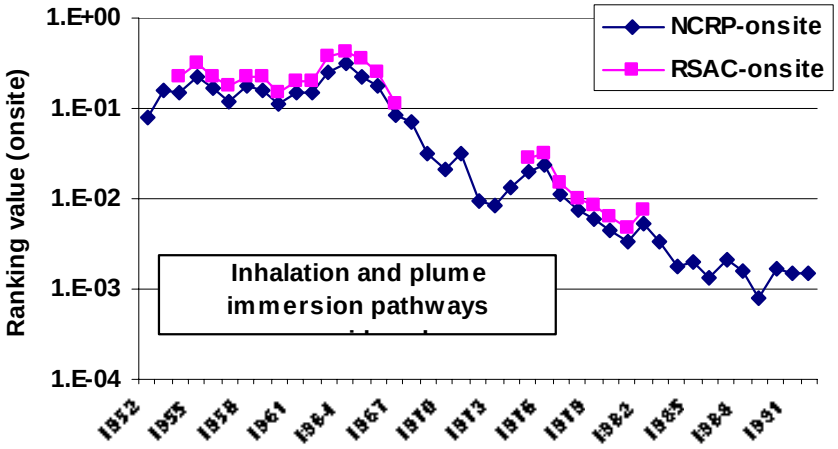


Figure 19. Results of ranking individual years at the *onsite* location from all radionuclides by the inhalation and plume immersion exposure pathways. 1963, 1964, and 1965 are the years with highest ranking values at the *onsite* location. The RSAC program, used to estimate ranking values for 1954-1966, confirms the NCRP ranking results.

For the important years, the radionuclides with the highest ranking values during those years were similar to those radionuclides that emerged when all years were considered (see Tables 7 and 8). For example, ranking releases during the RaLa processes, which occurred during the high release years of 1957, 1958 and 1959, showed the important radionuclides were still ^{131}I, ^{90}Sr, and ^{137}Cs. (Table 9a). The effect of the releases from the RaLa runs was seen onsite as well, with ^{131}I

and ^{90}Sr ranking highest in 1957 (Table 9b). In contrast, ^{41}Ar, and several xenon and krypton isotopes were the top contributors when all years were considered at the onsite location (see Table 8). Nevertheless, ^{131}I and ^{90}Sr were still ranked high among the radionuclides when all years were evaluated. In summary, the radionuclides released to air from the INEEL that were the most important contributors to *offsite* exposures at Atomic City in 1957 were ^{131}I, ^{137}Cs, and ^{90}Sr. At the *onsite* exposure location, ^{131}I, ^{90}Sr, ^{41}Ar, and ^{144}Ce were the main contributors to exposure from the inhalation and plume immersion pathways.

Table 9a. Ranking Values for Exposure at the *Offsite* Location to Radionuclide Releases for All Pathways in 1957[a]

Radionuclide	Activity (Ci) 1957	Relative ranking value NCRP(offsite)	RSAC (offsite)	Ratio NCRP/RSCA
I-131	1.4E+03	2.76	0.63	4.4
Cs-137	3.6E+01	0.56	0.11	5.2
Sr-90	3.6E+01	0.48	0.21	2.3
Ce-144	5.4E+01	0.020	0.012	1.7
Ba-140	5.9E+01	0.010	0.001	7.8
I-133	4.4E+02	0.0068	0.0016	4.3
Zr-95	2.1E+01	0.0061	0.0035	1.8
Cs-134	6.1E-01	0.0056	0.0028	2.0
Pu-238	7.8E-02	0.0049	0.0056	0.9
Sr-89	1.9E+01	0.0049	0.0010	4.9
Ru-106	5.9E+00	0.0040	0.0013	3.1
Ar-41	3.4E+04	0.0036	0.0025	1.5
I-132	4.0E+03	0.0034	0.0047	0.7
Sb-125	2.9E+00	0.0033	0.0009	3.7
Pm-147	1.4E+02	0.0032	0.0041	0.8
Xe-138	1.8E+04	0.0028	0.0004	7.2
Y-91	2.1E+01	0.0027	0.0019	1.4

[a] Of over 50 radionuclides evaluated during 1957, these ranked highest at the offsite location.
[b] A ratio of one indicates agreement between the ranking values obtained from the NCRP and RSAC methods. These ratios indicate a tendency for the NCRP method to yield higher ranking values.

Table 9b. Ranking Results for the Inhalation and Plume Immersion Exposure Pathways to Radionuclide Releases at the *Onsite* Location in 1957 [a]

Radionuclide	Activity released (1957) Ci	Relative ranking value NCRP (onsite)	RSAC (onsite)	Ratio NCRP/RSAC
I-131	1.40E+03	0.047	0.035	1.3
Sr-90	3.57E+01	0.028	0.035	0.8
Ar-41	3.42E+04	0.014	0.021	0.7
Ce-144	5.40E+01	0.012	0.015	0.8
Pu-238	7.78E-02	0.011	0.023	0.5
Xe-138	1.83E+04	0.0071	0.0018	4.0
Xe-133	4.63E+05	0.0055	0.0086	0.6
Kr-88	5.48E+03	0.0044	0.0057	0.8
I-132	4.03E+03	0.0043	0.0054	0.8
Pm-147	1.40E+02	0.0032	0.0041	0.8
I-133	4.41E+02	0.0025	0.0020	1.2
Pu239,240	1.15E-02	0.0017	0.0037	0.5
Ru-106	5.87E+00	0.0016	0.0021	0.8
Kr-87	5.59E+03	0.0015	0.0019	0.8
Xe-135	1.83E+04	0.0015	0.0025	0.6
Cs-137	3.59E+01	0.0007	0.0009	0.8
Y-91	2.11E+01	0.0006	0.0008	0.8
Sr-89	1.92E+01	0.0005	0.0006	0.8
Xe-135m	3.28E+03	3.4E-04	1.30E-04	2.6

[a] Of over 50 radionuclides evaluated during 1957, these ranked highest at the onsite location for the inhalation and plume immersion pathways.

[b] A ratio of one indicates agreement between the ranking values obtained from the NCRP and RSAC methods. Ratios less than one indicate a tendency for the NCRP method to yield lower ranking values than the RSAC program.

Viewing the NCRP ranking values for radionuclides released to air for other years can illustrate the general decrease in radionuclide releases with time, and also can reflect the occurrence of episodic events during the year. As an example, Tables 10a and 10b show the ranking results for 1964 and 1975. In 1964, ^{106}Ru emerged as the highest ranked radionuclide along with ^{90}Sr, ^{137}Cs, ^{41}Ar, and ^{129}I at the *offsite* location. High releases of ruthenium occurred in 1964 from the WCF at the ICPP during an episodic event, which is evaluated as an episodic release. In 1975, radioactive releases to air were much lower than in previous years, which is reflected in the lower ranking values, compared to 1957 (Table 9a) or 1964 (Table 10a). Nevertheless, among the radionuclides that ranked highest in 1975 at the offsite location are some of the same radionuclides that are most important in 1964 (^{137}Cs, ^{90}Sr, and ^{129}I).

At the *onsite* location, the ranking values for radionuclides released in 1964 and in 1975 again show similar patterns with ^{41}Ar, ^{88}Kr, ^{138}Xe ranking high, although the overall ranking values are much lower in 1975. In 1964, the ruthenium release at the WCF is again reflected in ^{106}Ru ranking as one of the most important radionuclides at the *onsite* location, too.

As a final point, Tables 7, 8, 9a, 9b, 10a, and 10b, and Figures 18 and 19 confirm that the NCRP method was suitable for these ranking purposes when the results are compared with those using the RSAC code. In all cases, the RSAC code confirmed the results obtained using the NCRP methodology. The Excel spreadsheets, OffsiteAirRanking.xls and OnsiteAirRanking.xls, present all the ranking results for all years.

Table 10a. Radionuclides Ranked Highest from Releases to Air in 1964 and 1975 at the *Offsite* Location [a]

	1964-Offsite location			1975-Offsite location	
Radionuclide	Activity (Ci)	Ranking value	Radionuclide	Activity (Ci)	Ranking value
Ru-106	3.4E+02	0.23	Cs-137	6.0E-01	0.009
Sr-90	8.8E+00	0.12	Sr-90	2.4E-01	0.003
Cs-137	4.8E+00	0.074	Cs-134	3.5E-01	0.003
Ar-41	2.6E+05	0.027	I-129	7.4E-02	0.003
I-129	5.5E-01	0.022	Xe-138	1.6E+04	0.002
Xe-138	1.4E+05	0.021	Kr-87	1.0E+04	0.0007
C-14	6.1E+02	0.011	Ar-41	5.1E+03	0.0005
Co-60	7.2E-01	0.0086	Ru-106	7.9E-01	0.0005
Cs-134	9.3E-01	0.0085	Y-91	3.0E+00	0.0004
Ce-144	2.3E+01	0.0083	Xe-135	9.0E+03	0.0002
Sb-125	2.9E+00	0.0033	C-14	1.2E+01	0.0002
Y-91	2.4E+01	0.0031	Co-60	1.7E-02	0.0002
Kr-87	4.2E+04	0.0029	Na-24	2.4E+01	0.0002
I-131	1.4E+00	0.0027	Hg-203	9.0E-01	0.0001
Xe-135	5.2E+04	0.0014	I-131	5.6E-02	0.0001
Hg-203	1.1E+01	0.0013	Xe-135m	4.1E+03	0.0001
Cs-138	3.0E+03	0.00095	Pu-238	1.6E-03	0.0001
Zr-95	3.1E+00	0.00089	Sb-125	4.4E-02	0.00005
Rb-89	3.3E+03	0.00080	Te-132	9.0E-01	0.00005
Xe-135m	2.4E+04	0.00063	Eu-154	4.9E-03	0.00004
H-3	3.5E+03	0.00046	Ce-144	1.1E-01	0.00004
Te-132	7.3E+00	0.00037	I-132	4.8E+01	0.00004
I-132	3.8E+02	0.00032	Ba-139	1.1E+03	0.00004
Ru-103	3.7E+00	0.00026	Kr-85m	2.5E+03	0.00004
Na-24	2.7E+01	0.00021	Cs-138	1.1E+02	0.00003

[a] The NCRP Methodology was used to rank these radionuclides, considering all exposure pathways.

Table 10b. Radionuclides Ranked Highest from Releases to Air in 1964 and 1975 at the *Onsite* Location [a]

	1964-Onsite location			1975-Onsite location	
Radionuclide	Activity (Ci)	Ranking value	Radionuclide	Activity (Ci)	Ranking value
Ar-41	2.6E+05	0.11	Kr-88	8.0E+03	0.006
Ru-106	3.4E+02	0.09	Xe-138	1.6E+04	0.006
Xe-138	1.4E+05	0.05	Kr-87	1.0E+04	0.003
Kr-88	4.1E+04	0.03	Ar-41	5.1E+03	0.002
Kr-87	4.2E+04	0.012	Xe-135	9.0E+03	0.0007
Ce-144	2.3E+01	0.005	Xe-135m	4.1E+03	0.00042
Xe-135	5.2E+04	0.004	Pu-238	1.6E-03	0.00022
Xe-135m	2.4E+04	0.002	Ru-106	7.9E-01	0.00022
Cs-138	3.0E+03	0.002	Kr-85m	2.5E+03	0.00014
Rb-89	3.3E+03	0.002	Ba-139	1.1E+03	0.00011
Y-91	2.4E+01	0.0007	U-238 + D	1.3E-03	0.000093
Kr-85m	1.2E+04	0.0006	Y-91	3.0E+00	0.000092
Rb-88	2.1E+03	0.0005	Rb-88	4.2E+02	0.000091
I-132	3.8E+02	0.0004	Cs-138	1.1E+02	0.000085
Pu-238	1.4E-03	0.0002	Na-24	2.4E+01	0.000057
I-129	5.5E-01	0.0001	I-132	4.8E+01	0.000051
U-238 + D	1.3E-03	0.0001	Pu239,240	3.1E-04	0.000047
Cs-137	4.8E+00	0.000091	Ce-144	1.1E-01	0.000025
Co-60	7.2E-01	0.000090	Kr-85	2.4E+04	0.000019
Na-24	2.7E+01	0.000065	Xe-133	1.4E+03	0.000017
Kr-85	8.2E+04	0.000064	I-129	7.4E-02	0.000013
Xe-133	4.4E+03	0.000053	Cs-137	6.0E-01	0.000011
Xe-133m	4.8E+03	0.000049	Cs-134	3.5E-01	0.000010
I-131	1.4E+00	0.000046	Rb-89	1.1E+01	0.000006
Zr-95	3.1E+00	0.000044	Xe-133m	5.9E+02	0.000006

[a] The NCRP Methodology was used to rank these radionuclides, considering the inhalation and plume immersion pathways.

Ranking the Relative Importance of Facilities

Finally, we evaluated the main facilities at the INEEL for their relative ranking values for exposure at the offsite location. We calculated the total ranking value for each facility by year by summing the ranking values for individual radionuclides for that year. Figure 20 shows results for ranking selected years for three main facilities at the INEEL: ICPP, TRA, and ANL-W. The results show that the ICPP had the highest ranking values from 1953-1959, and 1962-1964. Athough 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 ranking values.

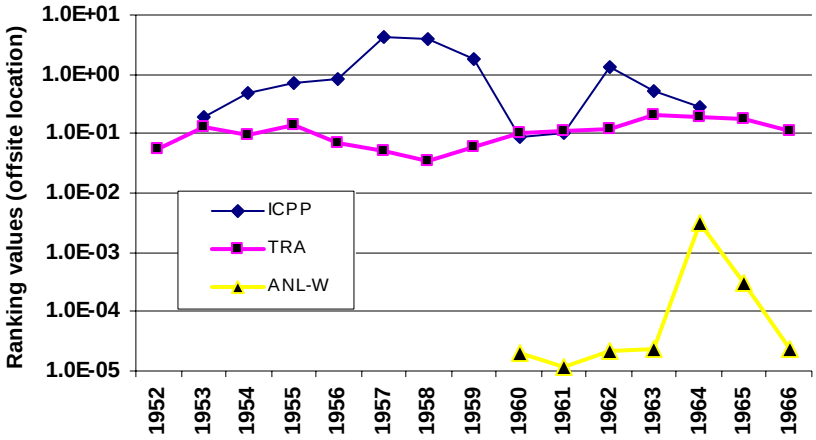


Figure 20. Comparison of total ranking values for radionuclide releases from the ICPP, TRA, and ANL-W from 1952 through 1964. The ICPP had the highest ranking values during most of this time period.

Table 11 summarizes information about main source of the radionuclides that ranked high in our screening assessment 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.

Table 11. INEEL Facility Source of Primary Radionuclides of Concern

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

EVALUATION OF INEEL EPISODIC RELEASES

Introduction

A number of government program 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 were the most important contributors to potential exposure for those events where multiple radionuclides were released. 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.

A preliminary dose reconstruction for the INEEL was published in 1991 when DOE (1991a) completed a Historical Dose Evaluation (HDE) to assess the dose impacts of both the routine and episodic releases that had occurred historically at the INEEL. This current task order project was undertaken with input and guidance from the INEEL Health Effects Subcommittee (HES) as an assessment of historic exposures to past releases from INEL that would be independent from DOE and the INEEL. The task order was designed by CDC to provide guidance and assistance in making decisions about what additional work, if any, may be needed to understand the impact of historical releases at the INEEL. We issued a draft report in September 2000 that was reviewed by the National Academy of Science committee. In the draft report, we made conservative but simplistic assumptions that used available and documented information about conditions existing at the time of the releases, and we also based some assumptions on work that had already been done as part of the HDE to compare and rank all episodic releases.

Based on review comments of the draft report, particularly those by the NAS committee, it was clear that this approach was considered to rely too heavily on the earlier work completed by DOE (1991a). In addition, the committee also was concerned that inconsistent levels of conservatism for the different releases had the potential to introduce biases in the ranking results. We agreed with their overall comments on the methodology and have modified the methodology somewhat for the final report. Reviewers recommended that we also more fully describe the use of NCRP screening factors, particularly for the episodic releases, because these factors were developed and intended for use with chronic or routine releases where the assumption of a constant annual average air concentration was more realistic. To this end, we have incorporated alternative methodologies as part of this final report to evaluate the efficacy of the NCRP screening factors for use in assessing the relative importance of short-term or episodic releases.

For this final report, we evaluated each episodic release using the air screening factors reported in NCRP (1996). As a comparison, we also evaluated a number of release events using other methods, including dose calculation made using the RSAC code and risk coefficients from EPA (Eckerman et al. 1999). The NCRP screening factors and models were not designed to assess episodic or short-term releases because of underlying assumptions within the model. However, we concluded the screening factors may still be viable tools for ranking and prioritizing the importance of episodic or short-duration releases. The additional comparisons with other methods are important to confirm or refute the efficacy of the screening factors for ranking the relative importance of release events (the NCRP screening factors are most readily applied, if they can be shown to be defensible for this purpose); to understand potential biases created by

using them; and to offer alternative methodologies for assessing episodic releases. The RSAC dose calculations and risk coefficients offer actual dose and risk values for understanding the potential magnitude of health impacts associated with the releases. Because the NCRP screening factors were not designed to estimate dose for short duration releases, we use them here to simply compute a ranking value that provides a way to compare one release to another. Based on the general agreement between the NCRP and RSAC methodologies, though, the NCRP air screening factors may be suitable in many cases for estimating doses related to short duration or episodic releases.

The final result is a different ranking for each methodology applied, and comparisons are made only among the releases within the five separate groups of releases, which are described in detail in a later section entitled Release Groups Established for this Evaluation. Because of differences in the assumptions made and degrees of conservatism for each group of release, we have not made comparisons across all episodic releases, nor do we make comparisons between episodic and routine releases. In addition to assessing releases at the nearest offsite receptor, we have also considered onsite exposure scenarios, which only incorporate the inhalation and plume immersion pathways.

Required Task

The task objective for this work was to list important releases of radioactive materials to the environment from these facilities and to rank those facilities whose releases warrant further study. *RAC*'s proposed approach for this work was to list, screen, and prioritize those incidents and operations at the INEEL, derived from the evaluation of the Phase I database, which have caused the release of radioactive material to the environment. To accomplish this, we proposed to rank the key radionuclides released to air or water from the INEEL as potential contributors to offsite radiation dose using the q methods developed by the National Council on Radiation Protection and Measurements (NCRP) (NCRP 1996). A ranking approach was considered appropriate because of the lack of existing models or methodologies designed to calculate meaningful screening dose estimates for short-term or episodic releases. Therefore, for this work, we proposed to evaluate past health impacts from all historical releases (routine and episodic) and complete a relative ranking of those releases with the level of effort and detail typically employed for a screening level approach.

However, because the description of the work to be performed as part of this project required a relative ranking of all release events and of the radionuclides comprising those releases, it was necessary to complete as realistic an assessment as possible for each release to avoid introducing inconsistent biases, which can significantly impact the results of the ranking. This is particularly true for shorter-term releases, where the specific meteorological conditions during and following the release greatly influence the potential impact of the release, which is highly dependent on the location of the potentially exposed individual.

Ideally, an iterative screening approach would be adopted for this task. For such an approach, releases are initially evaluated in a conservative but simplistic manner (i.e., an approach designed to avoid underestimating the impact of a given release) and measured (or screened) against a predetermined criterion (e.g., dose), and those releases that fall below the criterion are eliminated from further study without concern that a significant release has been overlooked. This type of screening process limits the number of releases for which more detailed

assessments may be necessary, makes the most efficient use of available resources, and provides an objective basis for making decisions about priorities for further study and for allowing researchers to focus on the radionuclides, facilities, programs, events, and time periods that are most important in terms of potential exposure to individuals or local populations. However, a criterion (e.g., specified dose level) against which release events could be evaluated (or screened) was not established in an effort to address CDC's desire to skip the controversial process of selecting an absolute (e.g., dose-based) decision criterion.

An evaluation of the INEEL episodic releases is also complicated by the fact that many of the events resulted in releases of unknown quantities of specific radionuclides. Therefore, the level of effort to derive a source term, or release amount, for each episodic release was considerably greater than is typically expected for a screening level assessment. An understanding of the potential health consequences of a given release, regardless of whether it is a screening level assessment or a more detailed evaluation, requires an estimate of the quantity of specific radionuclides that were released. Deriving estimates of these quantities involves reconstructing the operations that were responsible for the radionuclides that may have been present at the time of the release, as well as estimating the quantities of those radionuclides that may have actually been released. Furthermore, the level of detailed information available to understand these aspects of each release varies substantially for different release events, which exacerbates the potential for introducing inconsistent biases into the overall analysis. The following sections describe in detail the approach we have taken to produce a defensible ranking of the episodice release events that have occurred historically at the INEEL.

Selection Of Important Radionuclides

Because of the extremely large number of different radionuclides potentially released during the episodic releases that yielded fission products from reactor operations, accidents, or criticalities, it was important to identify those radionuclides most important in terms of health impacts to exposed individuals. This section discusses the process we used to select the radionuclides to consider for releases requiring complete reconstruction as part of this project (Group 1 and 2 releases).

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 constituents.

For this project, 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). DOE (1991a) evaluated the entire inventory produced as a result of the assumed reactor operations, whereas we fractionated the calculated inventory based on conservative release fraction estimates to more closely represent the composition of the actual material released. We also 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 different approach that allowed for evaluating the potential importance of the shorter-lived radionuclides to which members of the public may have been exposed.

We used the RSAC-5 computer code (Wenzel 1994) to calculate 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 12 and 13 show these two assumed operational scenarios along with, for comparison to the assumed scenarios, reactor operating parameters for actual transient operations and accidents (Table 12) and sustained operations (Table 13) 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 (including cesium and ruthenium isotopes, release fractions for which must be entered separately when using RSAC), 0.5 for halogens, and 1.0 for noble gases. These release fractions are based on experiments on fission product release from controlled melting of fuel speciments conducted by Parker and Creek at the Oak Ridge National Laboratory and are consistent with the release fractions assumed as part of the experimental design for fission product field release tests (Convair 1959). These release fractions are also consistent with the expectation that the release fraction for noble gases should be greater than or equal to the release fraction for halogens, and that the release fraction for halogens should be greater than the release

fraction for radioactive solids (DOE 1991a). We deleted all radionuclides with zero inventory at a decay time of 20 minutes, which is consistent with the shortest transit time for any of the episodic releases evaluated by DOE (1991a). This resulted in a list of 233 radionuclides.

Table 12. 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 13. 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 values were summed for all radionuclides present at a given decay time, and a percent contribution value for each radionuclide was computed. 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). They were originally designed to help facilities determine compliance and were not intended to be used for prioritizing different radionuclides, pathways, and sources. However, the screening models use certain fundamental principles of risk assessment that we believe can be adapted to help understand the relative importance of different sources of risk to focus efforts on the most important contributors to risk. It is emphasized that the calculation described above does not result in an estimate of dose; rather the intent is to identify those radionuclides that would be the most important to consider when evaluating releases consisting of a large number of different fission products, such as is the case for many of the INEEL releases.

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 their efficacy for this purpose further in Appendix B

We used 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 in the NCRP report. 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 screening value 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 screening value. 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 14 shows half-lives and maximum beta particle energies for the 10 radionuclides selected for this evaluation.

The relative screening values 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 15 lists estimated screening factors for the 10 radionuclides without existing screening factors that were selected for inclusion and evaluation in all episodic releases.

Table 14. 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 15. 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 13). 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 screening value. Beginning with the radionuclide with the highest percentage, we selected those radionuclides accounting for a cumulative 99.99% of the total relative screening value 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 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 releases requiring complete reconstruction of the source term (Group 1 and 2 releases). Releases consisting of known quantities of radionuclides not in this list were evaluated on a case-by-case basis.

Approach for Ranking INEEL Episodic Releases

A relative ranking of all release events was not achievable without a detailed and realistic assessment that considers the meteorological conditions at the time of the release, as well as the many different physical processes that govern the quantities of specific materials that may have been produced and released to the environment. This level of detail is not intended nor is it warranted at this stage in the evaluation of potential risk associated with a facility or release event, where simplistic and conservative assumptions are generally used to avoid unnecessary expenditure of time and resources. However, a relative ranking, if it is to be credible, requires some minimum level of detail and realism and cannot be based entirely on the simplistic and conservative assumptions typically used for screening level analyses.

One option we believe is defensible for developing a prioritized list of releases, without an established screening criterion, is to divide the episodic release events into several different groupings for which the level of available information and required assumptions are generally consistent for the releases within each grouping. While this approach limits the likelihood of introducing significant inconsistent biases that could impact the validity of the ranking results, it also means that a single ranked list of releases was not achievable within the scope of this work. However, we believe the rankings established for each grouping are defensible and can enable more focused work to be directed at those release events ranked highest within each group. We stress, however, that the ranking values established for a given episodic release are not directly comparable with ranking values established for other releases in a different grouping.

Release Groups Established for this Evaluation

Based on the above discussion, we established the following groupings for ranking the episodic releases falling within each group:

Group 1

This group consists of longer duration release events, requiring complete reconstruction of the composition of the release, and for which DOE (1991a) reported dispersion factors. Releases included in Group 1 are shown in Table 17.

Group 2

This group consists of shorter duration release events, requiring complete reconstruction of the composition of the release, and for which DOE (1991a) reported dispersion factors. Releases included in Group 2 are shown in Table 17.

Group 3

This group consists of release events for which known quantities of radionuclides were released (i.e., reconstruction of the release was not required) and for which DOE (1991a) reported dispersion factors. With the exception of the NRF S1W Engineering Test release, these are shorter duration release events.

Group 4

This group consists of release events occurring over the course of several days or more, for which known quantities of radionuclides were released, and that DOE (1991a) did not evaluate explicitly as episodic releases and therefore did not report dispersion factors.

Group 5

This group consists of release events occurring during a period of one day or less, for which known quantities of radionuclides were released, and that DOE (1991a) did not evaluate explicitly as episodic releases and therefore did not report dispersion factors.

Because the assumptions required and information available to characterize each release within a given group are similar, the potential for introducing inconsistent biases into the calculations that could impact the ranking within each group is significantly reduced. There are several key pieces of information that can significantly impact analysis of a given release, and therefore its position within a ranked list:

1) Atmospheric dispersion. The movement of material in the atmosphere is impacted by the meteorological conditions at the time of the release (e.g., wind speed and direction, atmospheric stability, and mixing depth), as well as such things as the effective height of the release.
2) Source term. The amount of material released to the atmosphere is related to a) the operations that generated fission products and other radionuclides that may have been present during the release, and b) the nature of the test or accident that resulted in some fractional release of those radionuclides. In addition to the total quantity of radioactivity released to the atmosphere, the composition of specific radionuclides comprising the release plays a significant role with regard to the potential health consequences associated with the release.
3) Timing and duration of the release. Some releases occurred during periods of the year when the ingestion pathway is important to consider, while other releases occurred during

> winter months when the ingestion pathway would not be considered a significant pathway of exposure. Additionally, some releases occurred over the course of tens of minutes, while others occurred over the course of tens of days.

Each of these factors was considered in determining how the various release events should be grouped to enable a defensible ranking, and the following sections provide additional discussion of the considerations made and the methodology used to address each factor.

Atmospheric Dispersion

For episodic releases, the meteorological conditions existing at the time of the release is one of the most important considerations in assessing the relative importance of a given release event by comparison to other releases with regard to potential health consequences to members of the public. The following sections discuss the approach we used to account for atmospheric dispersion for each episodic release.

Releases For Which Dispersion Factors Were Available (Groups 1, 2, and 3)

Typically, a screening analysis would assume uniformly simplistic and conservative wind speeds (e.g., 2 m s^{-1}) and stability classes (F – stable, resulting in minimal dispersion) and a direct trajectory to the nearest offsite location where exposures to members of the public could occur. Then, more realistic assumptions based on existing conditions at the time of the release would be made for only those events with screening level dose estimates above some predetermined dose criterion.

Because this approach was not amenable to the ranking of releases called for in the scope of work, it was necessary to consider the conditions existing during the time period of the release wherever possible. The relatively detailed and realistic assessment completed by DOE (1991a) presented an opportunity to consider release specific meteorological conditions as part of this screening level analysis for those release events that were evaluated by DOE (1991a) (i.e., Group 1, 2, and 3 releases). The following is noted in Volume I of DOE (1991a):

> Once estimates have been made for the quantities of airborne radioactivity released, the next step in evaluating potential offsite radiation doses is to calculate atmospheric transport and dispersion of radioactivity to locations outside the INEL. These calculations were performed by the Field Research Division of the National Oceanic and Atmospheric Administration (NOAA), located in Idaho Falls, using the MESODIF computer code.

The dispersion factors used and reported by DOE (1991a) were derived from calculations made by NOAA, independently of DOE or the INEEL. Additionally, inquiries with the Idaho Falls NOAA office indicated that meteorological data prior to April 1993 were not readily available in electronic format (Hukari 2002a). Because the dispersion factors reported by DOE (1991a) were calculated by NOAA, independent of DOE, and because a more sophisticated treatment of atmospheric dispersion for the many different episodic releases was not possible with the resources available for this project, we considered their use to be the best approach for

considering release specific conditions. These dispersion factors represent the highest calculated values considering the potentially historically inhabited offsite locations identified by DOE (1991a) in each of the 16 22.5-degree sectors circumscribing the INEEL Site (Figure 21). Because these dispersion factors account for release specific information including stability, mixing depth, and wind speed, as well as source parameters such as stack height, they eliminate many of the inconsistent biases that would be introduced by making uniformly conservative and simplistic assumptions.

Because DOE (1991a) reports dispersion factors for offsite locations only, and because the scope of work for this project was expanded to consider the HESC's desire for an assessment of potential onsite exposures, we developed a methodology to estimate an onsite dispersion factor. This methodology was used for the release events for which DOE (1991a) reported dispersion factors and for which we calculate an onsite ranking value (i.e., Group 1 and 2 releases). Using the Gaussian plume and dispersion coefficient equations from Turner (1994), we estimated atmospheric dispersion as a function of *release specific* wind speed, stability class, stack height, mixing depth, and downwind distance to develop a ratio of expected dispersion at two different distances. The two assumed distances were approximated for, 1) a direct trajectory from the release point to the offsite location reported by DOE (1991a), and 2) the nearest potential onsite exposure location along that trajectory (typically a highway crossing the Site). In addition to estimating an onsite dispersion factor, wind speed information was needed to estimate a transit time to the point of exposure so that short-lived radionuclide activity could be allowed to decay according to that transit time. Different approaches were taken to estimate the wind speed and stability class, depending on the type of release.

For longer duration releases (including the NRF S1W Engineering Test and all IET releases except IET-6, -12, -13, -16, -22, and –24, which were assumed to occur on a single day), average wind speed and stability class information was obtained for the time period over which atmospheric releases were reported to have occurred. In cases where this information was not available, the time period of releases to the atmosphere was based on information reported by DOE (1991a). As previously noted, historical meteorological data corresponding to the time of each release were not readily available in electronic format (Hukari 2002a). Therefore, 5 years (1997 through 2001) of hourly wind speed, wind direction, and stability class data from the LOFT weather tower were obtained from the NOAA office in Idaho Falls (Hukari 2002b). For all longer duration IET releases, these data were used as a surrogate to estimate the average wind speed and stability class corresponding to the time period of each release using only those data falling within the 22.5-degree sector in which the release was estimated to have traveled, based on the location with the highest dispersion factor reported by DOE (1991a). Data from the GRID III (GRI3) weather tower for the period 1994 through 1998 were also examined in the same fashion and comparison of the two data sets (Table 16) suggests that this provides a reasonable approximation of the weather conditions expected to exist during the time of each release. The GRI3 data were used to estimate the average wind speed corresponding to the time of release for the NRF S1W Engineering Test release.

Table 16. Average Wind Speed and Stability Class Based on 5 Years[a] of Hourly Data from the GRI3 and LOFT Weather Towers at the INEEL for Time Periods Corresponding to Longer Duration Releases

Test	Date of release		Wind speed (m s^{-1})		Stability class[a]	
	Start	End	GRI3	LOFT	GRI3	LOFT
NRF S1W	6/18/55	6/30/55	4.7	4.8	4	4
IET-3	2/11/56	2/24/56	2.8	2.5	4	4
IET-4	5/1/56	6/29/56	6.9	6.5	4	4
IET-6	12/18/56		c	c	c	c
IET-8	7/31/57	8/28/57	6.7	6.5	4	4
IET-10	12/20/57	3/6/58	1.8	1.7	3	4
IET-11	3/20/58	4/14/58	2.5	6.1	3	4
IET-12	4/30/58	5/6/58	c	c	c	c
IET-13	11/18/58	11/18/58	c	c	c	c
IET-14	4/17/59	5/19/59	4.3	3.4	4	4
IET-15	6/3/59	6/24/59	4.1	3.3	4	4
IET-16	7/28/59	10/9/59	c	c	c	c
IET-17	11/2/59	12/12/59	1.7	3.5	5	5
IET-18	1/6/60	2/7/60	2.4	2.3	4	4
IET-19	2/17/60	4/30/60	4.3	4.1	4	4
IET-20	5/14/60	6/10/60	4.6	4.3	4	4
IET-21	6/29/60	8/6/60	4.2	4.2	4	4
IET-22	8/12/60	8/25/60	c	c	c	c
IET-23	9/7/60	10/14/60	3.9	3.9	4	4
IET-24	10/17/60	10/26/60	c	c	c	c
IET-25	11/22/60	12/15/60	2.3	2.2	4	4
IET-26	12/23/60	3/30/61	3.4	2.6	4	4

[a] Surrogate data from 5-year period 1997-2001 (LOFT) and 1994-1998 (GRI3) (GRI3 refers to GRID III, shown in Figure 21)
[b] Stability class 1=A, 2=B, 3=C, 4=D, 5=E, and 6=F
[c] Because the release was assumed to occur on a single day, the average wind speed and stability class for the multi-day time period was not estimated

For shorter-duration Group 2 releases, wind speeds and atmospheric conditions at the time of the release were primarily estimated based on the information provided in the documentation available for each release. For seven Group 2 release events (IET-6, -12, -13, -16, -22, -24, and the FECF filter break), such information was not located. In those instances, wind speed was estimated based on the distance and transit time to the Site boundary reported by DOE (1991a), and stability class D was assumed because it maximized the estimated onsite dispersion factor. Information about the atmospheric stability existing during the BORAX-I excursion was also not located, and stability class E was assumed because it maximized the estimated onsite dispersion factor.

For Group 3 releases (except the NRF S1W release), wind speeds at the time of the release were primarily estimated based on the information provided in the documentation available for each release. If wind speed information was not located, it was estimated based on the distance and transit time to the Site boundary reported by DOE (1991a). Unless available documentation provided specific information, the stability class that resulted in the maximum onsite dispersion factor was assumed.

The mixing depth corresponding to the time of the release was estimated based on data reported by Clawson et al. (1989) for the morning and afternoon periods during spring, summer, autumn, and fall. For this analysis, we assumed spring occurs between March 21 and June 20, summer between June 21 and September 22, autumn between September 23 and December 21, and winter between December 22 and March 20. For longer-term episodic releases and shorter-term releases for which a time of release was not documented, the average of the morning and afternoon mixing depths was used for the season during which the release occurred. For shorter-term episodic releases for which a time of release was documented, the morning or afternoon mixing depth for the season during which the release occurred was used.

Releases For Which Dispersion Factors Were Not Available (Groups 4 and 5)

For releases not evaluated by DOE (1991a) and, therefore, for which dispersion factors were not reported (i.e., Group 4 and 5 releases), it was necessary to compute dispersion factors. We based our evaluations on existing meteorological conditions wherever possible; however, for many of these releases this information was not located in available documentation. Unless available documentation provided information about wind speed or stability, we assumed a wind speed of 2 m s^{-1} and stability class F for all Group 4 releases and stability class D for all Group 5 releases. The mixing depth corresponding to the time of the release was estimated based on data reported by Clawson et al. (1989) for the morning and afternoon periods during spring, summer, autumn, and fall. For longer-term episodic releases and shorter-term releases for which a time of release was not documented, the average of the morning and afternoon mixing depths was used for the season during which the release occurred. The release height was assumed to be either equal to the physical height of the stack or equal to zero for ground-level releases. We calculated dispersion factors using the basic Gaussian plume equation and the Pasquill Gifford dispersion coefficient equations provided by Turner (1994).

Source Term

The quantity of specific radionuclides released to the atmosphere during an episodic event must be estimated to evaluate the potential health consequences of the release. The amount of information available and assumptions necessary to derive this information for each release event are quite variable.

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 evaluate in terms of potential dose to exposed individuals. We used the known quantities directly to evaluate these 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 assess because the precise quantity and composition of the release are not known. Therefore, evaluating these releases requires reconstructing the operations leading up to the episodic event and estimating 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 the fraction of each radionuclide that may have been released to the environment. Many of these events also involved the release of short-lived radionuclides, so it is necessary to consider decay during transit from the release point to the location of exposure because the importance of released radionuclides to potential dose changed as the short-lived radionuclides decayed. Decay times for releases in this category were estimated based on transit times resulting from the assumed wind speed for the release and the approximate distance to the location where potential exposure was assessed (i.e., either onsite or offsite location).

To determine the composition of radionuclides present at the time of the release, fission product inventories were estimated based on documented reactor operating histories or energy release estimates. Unless available documentation suggested otherwise, we assumed release fractions of 0.1 for solids (including cesium and ruthenium isotopes), 0.5 for halogens, and 1.0 for noble gases. The rationale for assuming these release fractions was discussed previously in the Transient Operation section. Where available documentation provided information about the total activity released, the release fractions were adjusted in an iterative fashion (maintaining the same proportions of 0.1 for solids, 0.5 for halogens, and 1.0 for noble gases) until our calculated release equaled or slightly exceeded the reported total release (or estimated total release based on reported release rates). It is important to note that even in cases where halogens and noble gases are the only radionuclides released, radioactive decay following the release results in the creation of substantial quantities of non-halogen and non-noble gas (i.e., solids).

In addition to fission products, many of the releases potentially included fuel constituents (i.e., uranium isotopes) and argon-41. Uranium releases were estimated based on the amount of fuel present, the enrichment of that fuel, and the release fraction assumed for solid fission products.

Air passing through the core of a reactor during operation contains stable argon, which can be neutron activated to form ^{41}Ar (1.8-hour half life). The quantity of ^{41}Ar produced is dependent on the natural abundance of stable argon in air and the thermal neutron flux to which the air is exposed. We examined the methodology used by DOE (1991a) to estimate ^{41}Ar production and release and determined that it was sufficient to provide an adequate estimate for a screening level assessment; therefore, we based our assumed ^{41}Ar releases on the methodology reported by DOE (1991a).

For the IET releases, ^{41}Ar production was estimated based on the methodology developed by DOE (1991a) for production in the HTRE No. 1 core. DOE (1991a) calculated a production rate of 2.8 Ci MW-hr^{-1} based on a conservatively assumed power level of 20 MW for all operations above 200 kW, the volume of air in the reactor core, the percentage of argon in the air, the thermal neutron cross-section for argon, the thermal neutron flux in the reactor core, and the upper value of airflow through the core. A production rate of 2.8 Ci MW-hr^{-1} was assumed for all IET releases. For other reactor operations potentially resulting in ^{41}Ar production (SNAPTRAN-

2), the methodology used by DOE (1991b) was based on an assumed volume of air surrounding the reactor, an estimate of the amount of argon gas in that volume of air, and an estimate of the number of atoms of argon activated by neutrons escaping the reactor core.

Timing and Duration of the Release

As noted previously, some releases occurred over the course of tens of minutes, while others occurred over a period of several days or more. For the purpose of ranking release events evaluated by DOE (1991a), we divided the releases into two categories based on the length of time the NOAA-calculated puff trajectories for each release continued to impact the exposure location, which is indicated by the "hours modeled" information provided in Table B-2 of DOE (1991a). This information provided a logical division according to duration for the various releases (Table 17). To rank release events not evaluated by DOE (1991a), we divided the releases into two general categories: 1) releases occurring during a period of one day or less, and 2) releases occurring over the course of multiple days.

Some releases occurred during periods of the year when the ingestion pathway is important to consider, while others did not. While this does not impact the way in which releases are grouped for this analysis, it does impact the exposure pathways that are considered for each release event. For releases occurring between November and April, the ingestion exposure pathway was not considered. For releases occurring between May and October, the ingestion exposure pathway was included.

Evaluation Methodology

The work plan for this project called for using the NCRP screening factors to assess the *relative* importance of each release event, based on the known or estimated composition of the release. The NCRP air screening factors are designed to demonstrate compliance with environmental standards or other administratively set reference levels for releases of radionuclides to the atmosphere. They apply to intermittent or continuous releases of radionuclides to the environment during routine operation over a period of 30 years with exposure to the releases assumed to be during a one-year period of the last year. The 30-year period is used for build up of radionuclides in the soil. Although, the NCRP screening factors were not designed to evaluate episodic or short-term releases explicitly, we believed the NCRP screening factors could be used to examine the *relative* importance of each release event. Furthermore, an alternative methodology or precedent for such an evaluation of short-duration releases is lacking at this time. It is emphasized that the results of the NCRP screening methodology (or any screening approach) do *not* represent realistic dose estimates. This point is important to understand. The intent of these calculations is to estimate the *relative importance* of each release event and of the individual radionuclides comprising each release, and **not** to make realistic dose or risk estimates. Subsequent to this study, more detailed calculations can then focus on those release events and/or radionuclides determined to be most important by comparison to other similar release events.

Table 17. Duration of Various Episodic Releases[a]

Group 1	Hours modeled	Group 2	Hours modeled
IET-4[b]	1420	1959 ICPP criticality	3
IET-8	167	1961 ICPP criticality	5
IET-10[b]	834	Borax	8
IET-11	260	FEBT-A	3
IET-14	241	FEBT-B	4
IET-15[b]	297	FECF Filter Break	6
IET-17[b]	574	FPFRT-1	15
IET-18	275	FPFRT-2	9
IET-19[b]	670	FPFRT-3	27
IET-20	371	FPFRT-4	28
IET-21	289	FPFRT-5	17
IET-23	310	FPFRT-6	53
IET-25[b]	294	FPFRT-7	40
IET-26[b]	361	FPFRT-8	23
		FPFRT-9	56
		IET-3	35
		IET-6	8
		IET-12	4
		IET-13	6
		IET-16	33
		IET-22	2
		IET-24	1
		SL-1 accident	16
		SNAPTRAN-2	6
		SNAPTRAN-3	1
		SPERT-1	2
		SPERT-2	8
		SPERT-3	3

[a] Indicated by the number of hours modeled as reported in Table B-2 (DOE 1991a).
[b] Releases were divided into multiple periods, and the hours modeled represent the sum for all periods.

It is reasonable to challenge the legitimacy of applying the NCRP screening methodology to short-term releases. Therefore, we incorporated alternative approaches for comparison to the ranking results obtained using the NCRP screening factors to examine their efficacy for assessing the relative importance of a given release. We believe this comparison will be helpful to scientists in the future in selecting a valid approach for screening releases of radionuclides of short duration. These alternative methodologies are discussed in the next sections and have been incorporated into the evaluations for both Group 1 and Group 2 release events. Additionally, we provide a more detailed investigation of differences between the NCRP and RSAC methodologies related to specific exposure pathways and radionuclides in Appendix B.

The ranking values calculated for each release are correlated to the extent possible to actual dose for the RSAC methodology, using the options for ingestion dose calculations from an acute or short-term release. Because of the assumptions used for deriving the NCRP screening factors, it was not clear at the outset of this project that their use for episodic releases would result in a reliable estimate of dose. Based on the agreement between RSAC and NCRP, though, it appears the NCRP air screening factors may be viable in many cases to estimate dose for short-term releases using the methodology described below. The magnitude and specific meaning of the ranking values calculated for each release is not particularly relevant for this project in that the primary intent of the calculations is to produce a relative ranking order for each group of releases. However, the values should approximate dose in the case of RSAC, they appear to also approximate dose in the case of NCRP (based on the close agreement with RSAC), and they approximate risk in the case of the EPA method.

Offsite Exposure Scenarios

To assess the relative importance of offsite exposures for each Group 1 and 2 release, two separate methodologies are employed. Release-specific information, including wind speed, stack height, mixing depth, and release duration are incorporated into the calculations via the dispersion factors estimated for each release.

First, the RSAC program is used to estimate the quantity (q) of each radionuclide (based on assumed reactor operations, release fractions, and decay times) at the offsite exposure location. This value is then divided by the number of seconds in a year to normalize each episodic release to an annual release rate. The estimated dispersion factor (X/Q) is then multiplied by the radionuclide release rate to obtain a ground-level average air concentration at that location. This air concentration for each radionuclide is multiplied by the appropriate NCRP screening factor (SF) for each exposure pathway (i.e., inhalation, immersion in the plume, external exposure from the ground, and ingestion) to derive a relative ranking value (RRV). The ranking values for relevant (i.e., some releases do not consider the ingestion pathway) exposure pathways for each release are then summed to derive a total ranking value. The basic sequence of steps shown in Equation 13 illustrates how the relative ranking values are calculated for each radionuclide and exposure pathway.

$$RRV = q \div t \times CF_1 \times \frac{X}{Q} \times SF \times CF_2 \qquad (13)$$

where

RRV	=	relative ranking value (rem)
q	=	quantify of radionuclide present at the exposure location (Ci)
t	=	number of seconds in a year (3.1536×10^7 s)
CF_1	=	factor to convert Ci to Bq (3.7×10^{10} Bq Ci^{-1})
X/Q	=	atmospheric dispersion factor (s m^{-3})
SF	=	NCRP screening factor (Sv per Bq m^{-3})
CF_2	=	factor to convert Sv to mrem (100 rem Sv^{-1})

Second, the RSAC program is used to estimate a relative ranking value for each radionuclide, as well as a total ranking value for each release event for each exposure pathway so the relative importance of each radionuclide and release event can be assessed. We directly entered the same dispersion factors used for the NCRP methodology calculations, as shown in Equation 13, to ensure the same air concentrations formed the basis for subsequent dose calculations for each pathway. The RSAC results, which are not impacted by the assumptions used in developing the NCRP screening factors, are then compared to the results obtained using the NCRP screening factors.

Because the fission product inventory estimates were made using the RSAC program, which is also capable of calculating dose (used here as a relative ranking value) based on the release of that inventory (as well as other directly input radionuclides, such as ^{41}Ar and the uranium isotopes considered for several releases) and subsequent transfer through environmental media, it was determined that using RSAC to calculate relative ranking values would be the most efficient and practical methodology.

The ground exposure and ingestion pathways are not evaluated using the EPA risk coefficients (described below for the onsite analysis) as part of the offsite exposure scenario analysis because it was determined that the additional calculations necessary to estimate ground surface and food product concentrations were not appropriate at this stage. These calculations are incorporated into the assessments for these pathways made using the RSAC program, as described above.

By calculating relative ranking values using the two methodologies described above, the differences between the values result only from variations in the way exposure to the radionuclides is determined, based on the same air concentrations at the point of exposure. This is confirmed by the excellent agreement achieved when comparing the inhalation values resulting from both methodologies, which is discussed in detail in Appendix B. As additional assumptions and parameter values are necessary to estimate exposure through other pathways, more variation between the two methodologies becomes apparent.

Onsite Exposure Scenarios

To assess the relative importance of onsite exposures for each Group 1 and 2 release, three separate methodologies were considered. The onsite exposure scenarios considered the inhalation and immersion pathways only because we assumed that individuals at onsite locations were there for only limited periods of time, which would substantially limit or preclude exposure to radionuclides via other pathways.

First, the RSAC program is used to estimate the quantity (q) of each radionuclide (based on assumed reactor operations, release fractions, and decay times) at the onsite exposure location. This value is then divided by the number of seconds in a year to normalize each episodic release to an annual release rate. The estimated dispersion factor (X/Q) is then multiplied by the radionuclide release rate to obtain a ground-level average air concentration at that location. This air concentration for each radionuclide is multiplied by the appropriate NCRP screening factor (SF) for the inhalation and immersion pathways only. The value for each pathway is summed to derive a total onsite ranking value for each release.

Second, a total integrated concentration (TIC) is calculated by multiplying the quantity of each radionuclide at the onsite exposure location in bequerels (Bq) by the estimated dispersion

factor. This TIC is then multiplied by EPA lifetime cancer morbidity risk coefficients (Eckerman et al. 1999) for inhalation and an assumed breathing rate (2.54×10^{-4} m^3 s^{-1}) to evaluate the relative importance of the inhalation pathway. The TIC is also multiplied by the risk coefficient for immersion to evaluate the relative importance of the immersion pathways. The ranking values obtained for each pathway are summed to derive a total onsite ranking value for each release, and for each radionuclide contributing to the release. The ranking results obtained using this methodology are compared to the results obtained using the NCRP screening factors. It is stressed that the relative ranked order suggested by each methodology is the desired result from these calculations, and the magnitude of the specific values calculated by the EPA methodology are not comparable with either the NCRP or RSAC methodologies.

Finally, as for the offsite exposure scenarios, the RSAC program is used to estimate a ranking value for each radionuclide, as well as a total ranking value for each release event for the inhalation and immersion exposure pathways so the relative importance of each radionuclide and release event can be assessed. The RSAC ranking results are then compared to the ranking results obtained using the NCRP screening factors and the EPA risk coefficients.

The three different methodologies are hereafter referred to simply as the NCRP, RSAC, and EPA methodologies for both the offsite and onsite evaluations.

Use of RSAC

RSAC-5 (Wenzel 1994) was used for the analysis we completed to select the most important radionuclides to be assessed for reconstructed reactor operations or criticalities that resulted in fission product production (Selection of Important Radionulcides). The radionuclide selection process was completed for the draft report and was not revised for this final report. Since the draft report was issued, a new version of RSAC was made available. Therefore, RSAC-6 (Wenzel and Schrader 2001) was used for this final report to reconstruct the episodic releases discussed for the Group 1 and 2 evaluations and also for the calculations made to compare the RSAC and NCRP methodologies. The program is hereafter refered to simply as RSAC.

Because RSAC was used to calculate ranking values it is important to document the assumed parameter values. We ran the code primarily using the default parameters that control distribution of radionuclides through the food chain. We did, however, correlate a number of values to correspond with the assumptions made for the NCRP screening factors. First, we assumed a breathing rate of 2.54 m^3 s^{-1} to correspond to the 8000 m^3 y^{-1} assumed for the NCRP screening factor development. Next we used a deposition velocity of 0 m s^{-1} for noble gases and 0.01157 m s^{-1} for all other elements to correspond to the 1000 m d^{-1} assumed for the NCRP screening factors. To account for differences in release duration, we selected the option to calculate ingestion doses from an acute (short-term) release. We set the time period for crop exposure to contamination equal to the hours modeled values (described as the time required for the plume of each release to disperse so completely that it no longer made a significant contribution to the total air concentration) for each release (Table 17), as reported by DOE (1991a). The same dispersion factors used for the NCRP methodology were directly entered into the RSAC input file to ensure that the same air concentration at each location of exposure formed the basis for the ranking value calculations by both methods.

There are also a number of differences in parameter values assumed by RSAC and the NCRP screening factors that we have not attempted to correlate. The RSAC program does not

incorporate a soil ingestion exposure pathway, whereas the NCRP ingestion screening factors do, which results in a higher NCRP ranking value in some cases. RSAC assumes a physical weathering half-life of 0.0021 h^{-1}, while NCRP assumes a value of 0.003 h^{-1}. We set the harvest duration time period to 7 days for all pathways, while the NCRP methodology assumes a period of 1 day for vegetables, 2 days for milk, and 7 days for meat. There are a number of other differences, including variations in the element specific transfer factors, dose conversion factors, assumed chemical form of each radionuclide, assumed buildup time in the soil, and treatment of the specific decay schemes for each radionuclide, that also have some impact on the resulting ranking values calculated as part of this project. However, we do not believe these differences make a significant impact on the overall ranking results, as demonstrated by the general agreement between the two methods for the Group 1 and 2 releases.

There is generally quite good agreement between the two methodologies, suggesting that the NCRP screening factors are suitable for understanding the **relative** importance of different releases and in most cases appear to provide a reasonable estimate of potential dose by pathway, based on a time integrated air concentration. This agreement is particularly true for releases consisting of a broad mixture of different radionuclides arising from fissioning in a reactor or as a result of a criticality event. There are some specific radionuclides, though, that show significant differences between the NCRP and RSAC methodologies. Time and resource limitations precluded an in-depth investigation into the specific causes of all differences between the RSAC and NCRP results; however, this issue is explored further in Appendix B.

Episodic Release Evaluations

Based on the methodology described above, cumulative ranking values for each release and ranking values for each radionuclide comprising the release for each pathway are calculated. For each release, ranking values are calculated 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. The offsite exposure locations used for this analysis are shown in Figure 21. Most of these locations represent likely occupied residences during the times of the majority of the releases. Three of the locations, however, probably do not represent occupied residences (DOE 1991a). 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. Onsite locations were chosen based on the first publicly accessible highway crossed by the plume as it traveled in an assumed straightline trajectory from the release point to the offsite exposure location.

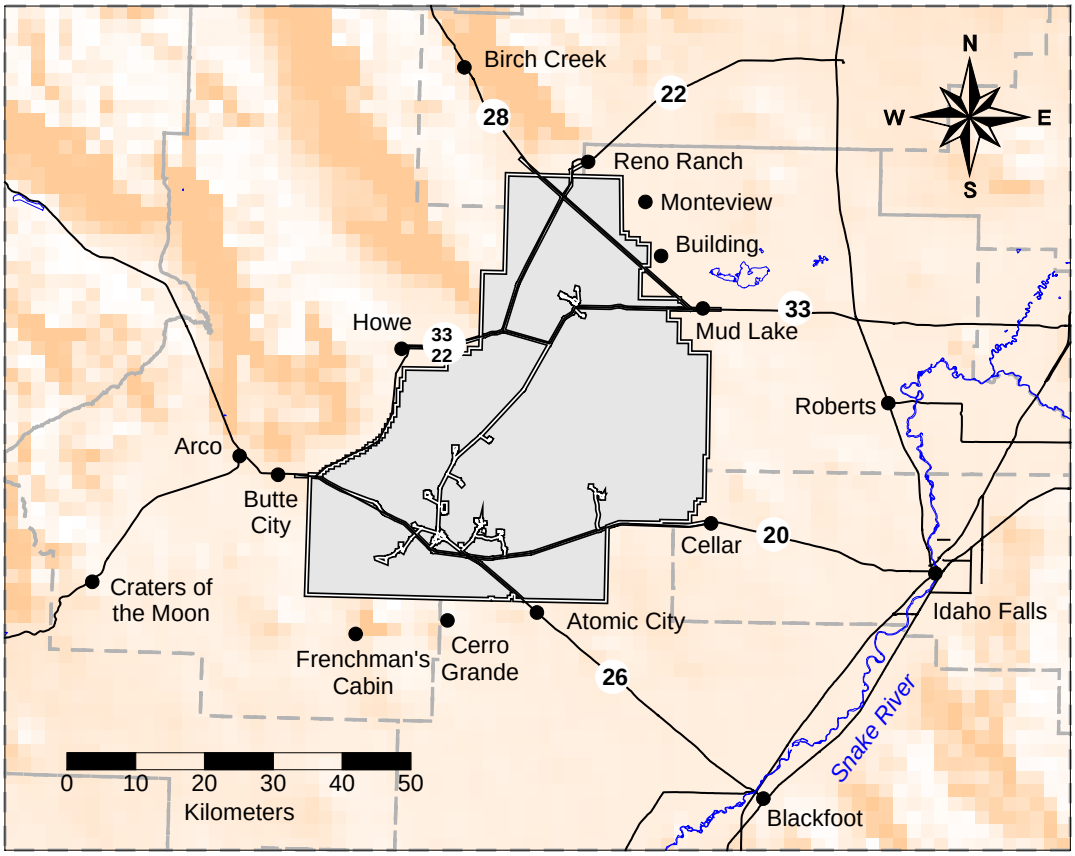


Figure 21. Map showing the 16 offsite locations at which potential exposure was assessed.

Supporting Information

Group 1 and 2 Releases

A number of supporting files are provided on the CD-ROM accompanying this report for each Group 1 and 2 release. All files related to each release are organized within a single folder, as illustrated in Figure 22.

The following provides a description of the supporting files, using the files for the IET-4 release as an example, as shown in Figure 22:

- For each release, the RSAC input and output files are provided. The output file contains the radionuclide inventory calculated for the onsite and offsite locations, as well as the RSAC dose (ranking value) calculations for the onsite and offsite locations.
  - input file: iet4
  - output file: iet4.out
- For all IET releases, one additional set of RSAC input and output files are provided. These additional files were used to estimate the total fission product release, which was correlated to the reported gross activity release or release rate.
  - input file: iet4(total)
  - output file: iet4(total).out

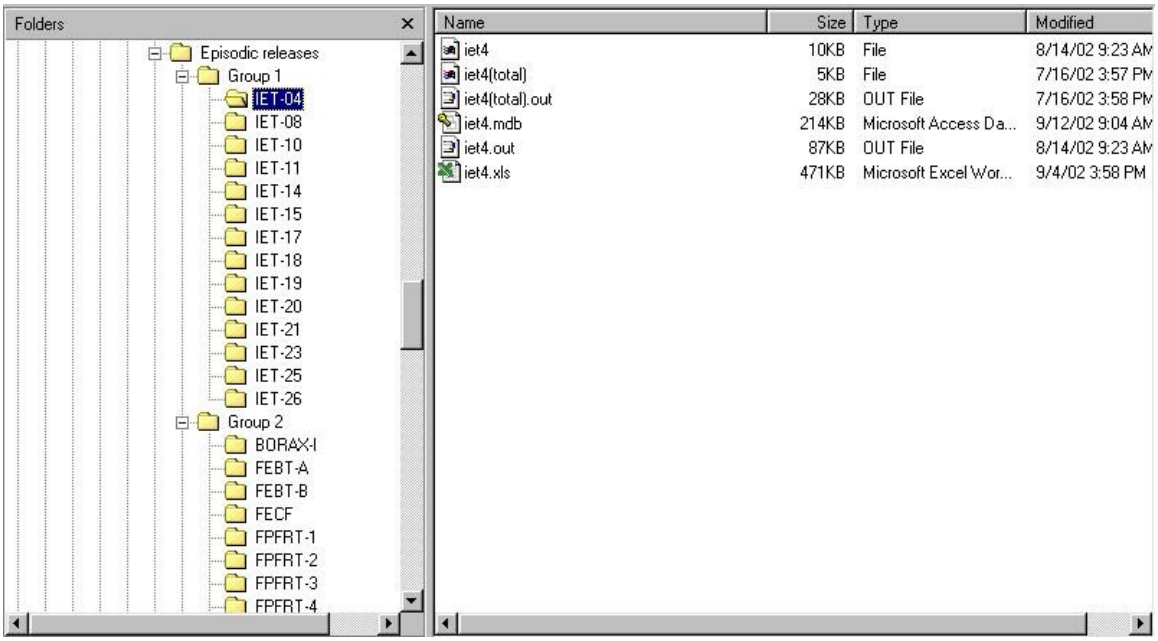


Figure 22. Directory structure for Group 1 and Group 2 episodic release supporting files. The files associated with the IET-4 release are shown here.

- A Microsoft Excel file is provided for each release (e.g., iet4.xls). This file contains the following spreadsheets:
  - Reactor history: Contains the reactor operating history, if it was compiled in Excel format; in some cases, reactor histories were directly input to the RSAC input file.
  - Uranium release: Shows the calculations to determine the activities of ^{234}U, ^{235}U, and ^{238}U estimated to be released, if applicable
  - Inventory: Shows the radionuclide activity expected at decay times selected to correspond with transit time to locations of exposure. This is the RSAC-calculated inventory.
  - RSAC (onsite): Shows the RSAC-calculated ranking values for the onsite location, if applicable, in a format suitable for import into a Microsoft Access table
  - RSAC (offsite): Shows the RSAC-calculated ranking values for the offsite location, in a format suitable for import into a Microsoft Access table
  - Onsite ranking values: Shows the NCRP, EPA, and RSAC ranking values for each radionuclide for both the inhalation and immersion pathways and computes a total onsite ranking value for each release
  - Onsite %: Shows the percent contribution by each radionuclide to the total onsite ranking value for each pathway. Yellow highlighted radionuclides are those for which NCRP screening factors do not exist and were estimated (discussed previously).

  - Offsite ranking values: Shows the NCRP and RSAC ranking values for each radionuclide for each pathway and computes a total offsite ranking value for each release
  - Offsite %: Shows the percent contribution by each radionuclide to the total offsite ranking value for each pathway. Yellow highlighted radionuclides are those for which NCRP screening factors do not exist and were estimated (discussed previously).
- A Microsoft Access file is provided for each release (e.g., iet4.mdb). This file contains the following tables:
  - Chi/Q: This table contains the estimated onsite and offsite dispersion factors for each release.
  - EPA risk coefficients: Compiles the EPA lifetime cancer morbidity risk coefficients for the inhalation and immersion pathways for each radionuclide where a coefficient is available.
  - Inventory (Ci): Contains the same data as the "Inventory" spreadsheet in the Excel file.
  - NCRP screening factors: Compiles the NCRP air screening factor for each radionuclide.
  - Radionuclides: Lists the radionuclides considered for each release
  - RSAC offsite: Contains the same data as the "RSAC (offsite)" spreadsheet in the Excel file.
  - RSAC onsite: Contains the same data as the "RSAC (onsite)" spreadsheet in the Excel file.

  and queries:
  - qryOffsite: Computes the offsite ranking values for each radionuclide and each pathway. The output of this query is pasted into the "offsite ranking values" spreadsheet in the Excel file.
  - qryOnsite: Computes the onsite ranking values for each radionuclide and each pathway. The output of this query is pasted into the "onsite ranking values" spreadsheet in the Excel file.
  - qryTIC: Calculates the total integrated air concentration used by the above two queries.
  - These queries ensure that the same calculations are completed for each release, using the same set of screening factors and risk coefficients, and limit the possibility of introducing errors into the calculations.

Group 3, 4, and 5 Releases

A number of supporting files are provided on the CD-ROM accompanying this report for the Group 3, 4, and 5 releases. All files related to each group are organized within a single folder. The following supporting files are provided for each group:

- A Microsoft Excel file for each group (e.g., Group 4.xls) containing the following spreadsheets, using the Group 4 file as an example:

  - Group 4 releases: Compiles the specific release quantities for each radionuclide for each release, along with onsite and offsite dispersion factors. This information is used to calculate an estimated average air concentration for each radionuclide.
  - Input to Access: Reduces the information in the above-described spreadsheet to just the average air concentration values, which are then imported into the Access file described below.
  - Ranking results: Shows the ranking values for each release, sorted from highest to lowest, based on the offsite values.
- A Microsoft Access file for each group (e.g., Group 4.mdb). This file contains the following tables:
  - NCRP screening factors: Compiles the NCRP air screening factor for each radionuclide.
  - Release information: Compiles the information from the "Input to Access" Excel spreadsheet described above.

  and queries:
  - qryRanking values: Computes an onsite and offsite ranking value for each radionuclide of each release, based on the average air concentration estimated for each location and the appropriate NCRP air screening factor.
  - qryGroup 4 ranking: Sums the ranking values calculated for each radionuclide of each release by the above query to derive an onsite and offsite total ranking value for each release. The output of this query is pasted into the "Ranking results" Excel spreadsheet, described above.

In addition to the above-described Microsoft Excel and Access files, RSAC files that compute the decay during transit for the NRF S1W Engineering Test release is provided in the Group 3 release folder.

Group 1 Evaluation

As discussed previously, this group consists of longer duration release events, requiring complete reconstruction of the composition of the release, and for which DOE (1991a) reported dispersion factors. Tables 18a and 18b show release-specific variables for each release event in this group. At the end of this section, relative ranking values are given for each release event.

Table 18a. Release-specific Variables for Group 1 Release Events

Release event	Dates of release		Hours modeled[a]	Season[b]	Ingestion[c]	Assumed mixing height (m)	Assumed wind speed[d] ($m\ s^{-1}$)	Assumed stability class[d]
	start	stop						
IET-4	5/1/56	6/29/56	1420	spring		1405	6.5	D
IET-8	7/31/57	8/28/57	167	summer		1580	6.5	D
IET-10	12/20/57	3/6/58	834	winter	no	565	1.7	D
IET-11	3/20/58	4/14/58	260	spring	no	na	na	na
IET-14	4/17/59	5/19/59	241	spring		1405	3.4	D
IET-15	6/3/59	6/24/59	297	spring		1405	3.3	D
IET-17	11/2/59	12/12/59	574	autumn	no	na	na	na
IET-18	1/6/60	2/7/60	275	winter	no	565	2.3	D
IET-19	2/17/60	4/30/60	670	winter/spring	no	985	4.1	D
IET-20	5/14/60	6/10/60	371	spring		1405	4.3	D
IET-21	6/29/60	8/6/60	289	summer		1580	4.2	D
IET-23	9/7/60	10/14/60	310	summer/autumn		1260	3.9	D
IET-25	11/22/60	12/15/60	294	autumn	no	940	2.2	D
IET-26	12/23/60	3/30/61	361	winter	no	565	2.6	D

[a] Source: DOE (1991a), Table B-2

[b] Spring: March 21 - June 20, Summer: June 21 - September 22, Autumn: September 23 - December 21, Winter: December 22 - March 20 (this seasonal division is approximate and some releases may overlap into another season by a few days)

[c] If release occurs between November and April, assumed no ingestion pathway contribution

[d] Wind speed and stability based on 5-year average at LOFT tower corresponding to specified time period of release

na – mixing height, wind speed, and stability estimates were not necessary because the onsite exposure scenario was not applicable

Table 18b. Release-specific Variables for Group 1 Release Events

			Offsite				Onsite				
Release event	Release Location	Stack height (m)	Location[a]	Downwind distance (km)	Transit time (minutes)	X/Q[b] (s m^{-3})	Location	Downwind distance (km)	Transit time (minutes)	X/Q ratio (offsite/onsite)	X/Q[c] (s m^{-3})
IET-4	TAN	46	Monteview	16	41	9.00E-09	HW 28	10	26	0.52	1.73E-08
IET-8	TAN	46	Monteview	16	41	1.00E-08	HW 28	10	26	0.52	1.93E-08
IET-10	TAN	46	Howe	25	245	1.92E-08	HW 22	9	88	0.24	7.98E-08
IET-11	TAN	46	Cellar	37	101	5.33E-09	na	na	na	na	na
IET-14	TAN	46	Frenchman's Cabin	54	265	7.63E-09	HW 20	39	191	0.64	1.19E-08
IET-15	TAN	46	Frenchman's Cabin	54	273	8.68E-09	HW 20	39	197	0.64	1.36E-08
IET-17	TAN	46	Cellar	37	176	3.96E-08	na	na	na	na	na
IET-18	TAN	46	Cerro Grande	45	326	1.97E-08	HW 20	38	275	0.79	2.49E-08
IET-19	TAN	46	Frenchman's Cabin	54	220	4.03E-08	HW 20	39	159	0.64	6.30E-08
IET-20	TAN	46	Blackfoot	72	279	6.30E-09	HW 20	32	124	0.33	1.91E-08
IET-21	TAN	46	Butte City	49	194	1.26E-08	HW 22/33	10	40	0.11	1.14E-07
IET-23	TAN	46	Monteview	16	68	1.21E-08	HW 28	10	43	0.52	2.33E-08
IET-25	TAN	46	Frenchman's Cabin	54	409	2.67E-08	HW 20	39	295	0.64	4.17E-08
IET-26	TAN	46	Frenchman's Cabin	54	346	2.43E-08	HW 20	39	250	0.65	3.73E-08

[a] See Figure 21

[b] Source: Table B-2 of DOE (1991a)

[c] Estimated based on calculated ratio of offsite to onsite dispersion factors (X/Q)

na – onsite exposure scenario was not applicable because the trajectory of the release did not result in the plume passing over an onsite publicly accessible highway

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. We provide some discussion related to each test series as part of this evaluation; however, the reader is encouraged to review the cited references for additional information related to the overall ANP program and the specific IET series comprising it.

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).

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). In particular, this is the case for the atmospheric dispersion factors and assumptions about the duration and timing of IET releases, as well as calculations to determine potential argon-41 generation. However, in all cases, we carefully evaluated the procedures used and assumptions made to ensure an appropriate evaluation.

To determine the composition of radionuclides present at the time of each IET release, fission product inventories were estimated based on documented reactor operating histories. The

actual release was based on available documentation about the total activity released or maximum reported release rates. Stack effluent measurements for IET-3, -4, and -6 were reported in terms of particulate activity only; however, methods were developed for later tests to relate the sample measurements to the total fission product release. We took this into consideration as part of the release fraction selection. The release fractions were adjusted in an iterative fashion (maintaining the same proportions of 0.1 for solids, 0.5 for halogens, and 1.0 for noble gases) until our calculated release equaled or slightly exceeded the reported total release (or estimated total release based on reported release rates). Uranium isotope and ^{41}Ar releases were estimated based on the methodology described previously in the section entitled "Source Term".

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. An important simplifying assumption that was made for all tests relates to the fact that the inventory estimated for each test, and consequently the estimated release, are based on the fission product composition existing at the end of the reactor operations for each test. In reality, the releases occurred during reactor operations; however, the scope of this task did not support this type of detailed investigation for each release.

Only the IET releases that fall into Group 1 are discussed and evaluated in this section. The IET releases falling into Group 2 are discussed and evaluated in that section.

IET-4

The IET-4 test series was conducted between April 17 and June 29, 1956, and DOE (1991a) assumed that atmospheric releases occurred between May 1 and June 29, 1956. The testing employed the HTRE-1 reactor assembly and the 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.

Particulate release rates (10-minute decayed) 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 h^{-1} release rate during the entire 192 hours of operation above 200 kW yielded a total particulate release of 35,700 Ci. We used RSAC to calculate a total fission product inventory, based on the operational history provided by Thornton et al. (1962). We iteratively adjusted the release fractions until our reconstructed operation resulted in a total 10-minute decayed particulate release of approximately 38,180 Ci (i.e., to derive a total release equal to or slightly exceeding the estimated gross activity release). The resulting release fractions were 0.0022, 0.011, and 0.022 for solids, halogens, and noble gases, respectively.

Thornton et al. (1962) reported that the total uranium (93.4% enriched) in the HTRE-1 core was 90 pounds (40,800 grams). We assumed the uranium in the core was released in the same

fraction used for the solids and estimated production and consequent release of 5782 Ci ^{41}Ar (2.8 Ci MW-h^{-1} times a total power generation of 2065 MW-h).

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, we assumed that radionuclide releases to the environment were negligible relative to releases associated with other reactor power operation tests.

IET-8

Evans (1957) reported that the IET-8 test series was performed between July 18 and August 28, 1957, and DOE (1991a) assumed that atmospheric releases occurred between July 31 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 a total of 33.97 hours of operation at maximum power had been accumulated during the test series. Additionally, no fission fragment evolution was measured, but the presence of molybdenum, manganese, and other radioactive particles was indicated by detection on filter papers.

We used RSAC to calculate an estimated fission product inventory, based on the operational history provided by Evans (1957) and further refined by DOE (1991a). 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. Evans (1957) states "Upon disassembly of the reactor, preliminary examination has shown molybdenum and manganese deterioration of the outer cladding of insert tube 3. Consequently, we assumed release fractions of 0, 0, and 1.0 for solids, halogens, and noble gases, respectively, in a single fuel tube (cartridge). In terms of the total core inventory, which consisted of 37 fuel cartridges, this equated to overall release fractions of 0, 0, and 0.027 for solids, halogens, and noble gases, respectively.

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, radioactive isotopes molybdenum and manganese 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 assumed the release consisted entirely of ^{93}Mo, the radionuclide with the highest total NCRP screening factor.

We assumed no release of uranium fuel constituents, which was consistent with the release of no particulate fission products. We estimated production and consequent release of 2742 Ci ^{41}Ar (2.8 Ci MW-h^{-1} times a total power generation of 979.4 MW-h).

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, we 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) reported that the IET-10 test series commenced on December 12, 1957 and was terminated on March 6, 1958. DOE (1991a) assumed that atmospheric releases occurred between December 20, 1957 and 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. Very little was known, at the time of this test, about the operational characteristics of ceramics. The manufacturing and design analysis techniques were likewise in early stages of development (Flagella 1962). 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. Foster et al. (1958) did not report the total power for Phase III, but DOE (1991a) reported 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 and RSAC to calculate a fission product inventory for the entire core. It appears likely that DOE (1991a) estimated the portion of total reactor power produced by the insert, which was reported to be 7.4% by Foster et al. (1958) and 10% by Flagella (1962).

There were three separate phases of testing during IET-10. Foster et al. (1958) reported 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 assumed the release rate reported for the previous run. Applying the corresponding reported rates of release during the 159 hours of assumed operation for the three phases of testing results in a total release of over 6.7×10^6 Ci. We calculated a total fission product inventory based on these assumptions, and iteratively adjusted the release fractions until our reconstructed operation resulted in a total 17-second decayed release of 6.8×10^6 Ci. The resulting release fractions were 0.075, 0.375, and 0.75 for solids, halogens, and noble gases, respectively.

Flagella (1962) reported that the total uranium (93.4% enriched) in the HTRE-2 core was 95 pounds (43,100 grams). We assumed the uranium in the core was released in the same fraction used for solids and estimated production and consequent release of 7048 Ci ^{41}Ar (2.8 Ci MW-hr^{-1} times a total power generation of 2517.2 MW-hr).

IET-11

The IET-11 test series was conducted between March 12 and April 14, 1958 (Evans 1958), and DOE (1991a) assumed that atmospheric releases occurred between March 20 and April 14, 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. There was no indication, however, of fuel cartridge damage. The unexpected fission product release was probably the result of uranium oxide, deposited in the lower cocoon during the insert 2B operation (IET-10), fissioning in the neutron-flux field below the reactor core (Flagella 1962).

Flagella (1962) reported 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 results in a total release of 1978 Ci. We used RSAC to calculate an estimated fission product inventory, based on the operational history for IET-11 provided by Evans (1958). Although the release may have resulted from fissioning of uranium oxide deposited during IET-10, an inventory derived based on the IET-11 operational history should provide a reasonable approximation of the composition of the fission product inventory available for release. We adjusted the release fractions until our reconstructed operation resulted in a total 10-minute decayed release of approximately 2110 Ci. The resulting release fractions were 0.00006, 0.0003, 0.0006 for solids, halogens, and noble gases, respectively.

Flagella (1962) reported that the total uranium (93.4% enriched) in the HTRE-2 core was 95 pounds (43,100 grams). We assumed the uranium in the core was released in the same fraction used for solids and estimated production and consequent release of 3675 Ci ^{41}Ar (2.8 Ci MW-h^{-1} times a total power generation of 1312.5 MW-h).

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 through May 19, 1959 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, in the HTRE-2 reactor assembly. 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. Pincock (1959) notes: "Some crystal deposits were formed on all fueled stages and on the upstream end of stage 11. The heaviest deposits were towards the center of the fueled region and on stages 7 and 8."

Pincock (1959) and Boone et al. (1959) reported a gross fission product release of 8656 Ci of 10-minute decayed activity. We used RSAC to calculate an estimated fission product inventory, based on the operational history provided by Pincock (1959). We adjusted the release fractions until our reconstructed operation resulted in a total 10-minute decayed release of approximately 8900 Ci. The resulting release fractions were 0.0002, 0.001, 0.002 for solids, halogens, and noble gases, respectively.

Flagella (1962) reported that the total uranium (93.4% enriched) in the HTRE-2 core was 95 pounds (43,100 grams). We assumed the uranium in the core was released in the same fraction used for solids and estimated production and consequent release of 3799 Ci ^{41}Ar (2.8 Ci MW-h^{-1} times a total power generation of 1356.7 MW-h).

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 employed the HTRE-2 reactor assembly to evaluate 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. Following the test, post-operational examination showed no damage to the outer-most fuel sheet of the cartridge, with inner sheet blisters, ruptured in some cases, on several stages (Evans 1959).

Boone et al. (1959) reported a gross fission product release of 899 Ci of 10-minute decayed activity. We used RSAC to calculate an estimated fission product inventory, based on the operational history provided by Evans (1959). We adjusted the release fractions until our reconstructed operation resulted in a total 10-minute decayed release of approximately 933 Ci. The resulting release fractions were 0.00003, 0.00015, 0.0003 for solids, halogens, and noble gases, respectively.

Flagella (1962) reported that the total uranium (93.4% enriched) in the HTRE-2 core was 95 pounds (43,100 grams). We assumed the uranium in the core was released in the same fraction used for solids and estimated production and consequent release of 2733 Ci ^{41}Ar (2.8 Ci MW-h^{-1} times a total power generation of 975.9 MW-h).

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. Post-operational examination of the fueled tubes showed no crystal growths or white powder deposits. The inside diameter of some of the tubes appeared to be rougher than others with variations in the degree of blackness also being noted. The outside surface of some of the tubes was streaked with a substance having a tan color (Evans 1960).

Pincock (1960a) reported a gross fission product release of 2017 Ci of 10-minute decayed activity. We used RSAC to calculate an estimated fission product inventory, based on the operational history provided by Pincock (1960a). We adjusted the release fractions until our reconstructed operation resulted in a total 10-minute decayed release of approximately 2100 Ci. The resulting release fractions were 0.000084, 0.00042, 0.00084 for solids, halogens, and noble gases, respectively.

Flagella (1962) reported that the total uranium (93.4% enriched) in the HTRE-2 core was 95 pounds (43,100 grams). We assumed the uranium in the core was released in the same fraction used for solids and estimated production and consequent release of 4177 Ci ^{41}Ar (2.8 Ci MW-h^{-1} times a total power generation of 1491.6 MW-h).

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. DOE (1991a) assumed that atmospheric releases occurred between January 6 and February 7, 1960. This test series was an extension of the IET-16 test series, designed to help define reactor operational parameters. Post-operational visual inspection of two fuel elements exhibiting higher temperatures than anticipated did not indicate any surface damage (Highberg et al. 1960).

Highberg et al. (1960) reported a gross fission product release of 1157 Ci of 10-minute decayed activity, as well as a maximum release rate of 8.6 Ci h^{-1}. Conservatively assuming an 8.6 Ci h^{-1} release rate during the entire 162 hours of operation yields a total release of 1394 Ci. We used RSAC to calculate an estimated fission product inventory, based on the operational history provided by Highberg et al. (1960) and further refined by DOE (1991a). We adjusted the release fractions until our reconstructed operation resulted in a total 10-minute decayed release of approximately 1450 Ci. The resulting release fractions were 0.00001, 0.0005, 0.0001 for solids, halogens, and noble gases, respectively.

GE (1959) reported that the total uranium (93.4% enriched) in the HTRE-3 core was 425 pounds (193,000 grams). We assumed the uranium in the core was released in the same fraction used for solids and estimated production and consequent release of 13,593 Ci ^{41}Ar (2.8 Ci MW-h^{-1} times a total power generation of 4854.6 MW-h).

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 the HTRE-2 reactor assembly and evaluated 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. Upon disassembly of the cartridge, visual examination showed the tubes to be in excellent condition. Some reddish color was noticed on the ends of the fueled tubes with the most intensive color appearing on the hottest stages (Pincock 1960b).

Pincock (1960b) reported a gross fission product release of 2908 Ci of 10-minute decayed activity. We used RSAC to calculate an estimated fission product inventory, based on the operational history provided by Pincock (1960b). We adjusted the release fractions until our reconstructed operation resulted in a total 10-minute decayed release of approximately 2990 Ci. The resulting release fractions were 0.00008, 0.0004, 0.0008 for solids, halogens, and noble gases, respectively.

Flagella (1962) reported that the total uranium (93.4% enriched) in the HTRE-2 core was 95 pounds (43,100 grams). We assumed the uranium in the core was released in the same fraction used for solids and estimated production and consequent release of 8610 Ci ^{41}Ar (2.8 Ci MW-h^{-1} times a total power generation of 3075.1 MW-h).

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 employed the HTRE-2 reactor assembly to evaluate 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. Post-operational examination showed a number of tubes to be cracked or broken throughout the fuel cartridge (Foster et al. 1960).

Foster et al. (1960) reported a gross fission product release of 5119 Ci of 10-minute decayed activity. We used RSAC to calculate an estimated fission product inventory, based on the operational history provided by Foster et al. (1960). We adjusted the release fractions until our reconstructed operation resulted in a total 10-minute decayed release of approximately 5280 Ci. The resulting release fractions were 0.00018, 0.0009, 0.0018 for solids, halogens, and noble gases, respectively.

Flagella (1962) reported that the total uranium (93.4% enriched) in the HTRE-2 core was 95 pounds (43,100 grams). We assumed the uranium in the core was released in the same fraction used for solids and estimated production and consequent release of 5126 Ci ^{41}Ar (2.8 Ci MW-h^{-1} times a total power generation of 1830.6 MW-h).

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 the HTRE-2 reactor assembly and evaluated 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. Descriptions of post-operational cartridge conditions were not provided (Pincock 1960c).

Pincock (1960c) reported a gross fission product release of 2688 Ci of 10-minute decayed activity. We used RSAC to calculate an estimated fission product inventory, based on the operational history provided by Pincock (1960c). We adjusted the release fractions until our reconstructed operation resulted in a total 10-minute decayed release of approximately 2740 Ci. The resulting release fractions were 0.00044, 0.0022, 0.0044 for solids, halogens, and noble gases, respectively.

Flagella (1962) reported that the total uranium (93.4% enriched) in the HTRE-2 core was 95 pounds (43,100 grams). We assumed the uranium in the core was released in the same fraction used for solids and estimated production and consequent release of 975 Ci ^{41}Ar (2.8 Ci MW-h^{-1} times a total power generation of 348.4 MW-h).

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, also used for IET-21, in the HTRE-2 reactor assembly. Descriptions of post-operational cartridge conditions were not provided (Pincock 1960c).

Pincock (1960c) reported a gross fission product release of 2125 Ci of 10-minute decayed activity. We used RSAC to calculate an estimated fission product inventory, based on the operational history provided by Pincock (1960c). We adjusted the release fractions until our reconstructed operation resulted in a total 10-minute decayed release of approximately 2210 Ci. The resulting release fractions were 0.00034, 0.0017, 0.0034 for solids, halogens, and noble gases, respectively.

Flagella (1962) reported that the total uranium (93.4% enriched) in the HTRE-2 core was 95 pounds (43,100 grams). We assumed the uranium in the core was released in the same fraction

used for solids and estimated production and consequent release of 1699 Ci ^{41}Ar (2.8 Ci MW-h^{-1} times a total power generation of 606.8 MW-h).

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. DOE (1991a) assumed that releases to the atmosphere corresponded to significant periods of operation between November 22 and December 15, 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 were not able to locate a description of the post-operational condition of the fuel elements, though DOE (1991a) noted that the IET-18 and IET-25 operations were quite similar. Based on reported gross activity releases, extensive damage was not apparent.

Highberg et al. (1961) reported a gross fission product release of 218 Ci of 10-minute decayed activity. We used RSAC to calculate an estimated fission product inventory, based on the operational history provided by Highberg et al. (1961) and further refined by DOE (1991a). We adjusted the release fractions until our reconstructed operation resulted in a total 10-minute decayed release of approximately 236 Ci. The resulting release fractions were 0.0000018, 0.000009, 0.000018 for solids, halogens, and noble gases, respectively.

GE (1959) reported that the total uranium (93.4% enriched) in the HTRE-3 core was 425 pounds (193,000 grams). We assumed the uranium in the core was released in the same fraction used for solids and estimated production and consequent release of 9181 Ci ^{41}Ar (2.8 Ci MW-h^{-1} times a total power generation of 3279.0 MW-h).

IET-26

The IET-26 test series was conducted between December 22, 1960 and March 31, 1961, and DOE (1991a) assumed releases to the atmosphere occurred between December 23, 1960 and March 30, 1961, when the reactor was operated at power levels above 130 kW. The test series employed the HTRE-2 reactor assembly to evaluate 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. Post-operational disassembly and examination showed the general appearance of all stages to be exceptionally good. There were no broken unfueled tubes. Blisters in the clad material were observed in the center tubes of stages 10 and 9 and to a somewhat lesser extent in stages 8 and 7. Almost the entire outside row of fueled tubes was broken in half or thirds in stages 10 through 6 (Field et al. 1961).

Field et al. (1961) reported 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 7189 Ci. We used RSAC to calculate an estimated fission product inventory, based on the operational history provided by Field et al.

(1961). We adjusted the release fractions until our reconstructed operation resulted in a total 10-minute decayed release of approximately 7420 Ci. The resulting release fractions were 0.00028, 0.0014, 0.0028 for solids, halogens, and noble gases, respectively.

Flagella (1962) reported that the total uranium (93.4% enriched) in the HTRE-2 core was 95 pounds (43,100 grams). We assumed the uranium in the core was released in the same fraction used for solids and estimated production and consequent release of 9104 Ci ^{41}Ar (2.8 Ci MW-h^{-1} times a total power generation of 3251.6 MW-h).

Group 1 Ranking Results

The following sections discuss the results of both the offsite and onsite ranking values for the Group 1 releases.

Offsite

Figure 23 provides a graphical representation of the offsite relative ranking values for each Group 1 release, sorted from highest to lowest according to the NCRP total ranking values (Table 20). The values obtained by the NCRP and RSAC methodologies are quite comparable, and both methodologies result in the same general relative ranking, with IET-10 and IET-4 releases clearly appearing as the highest-ranking releases in Group 1.

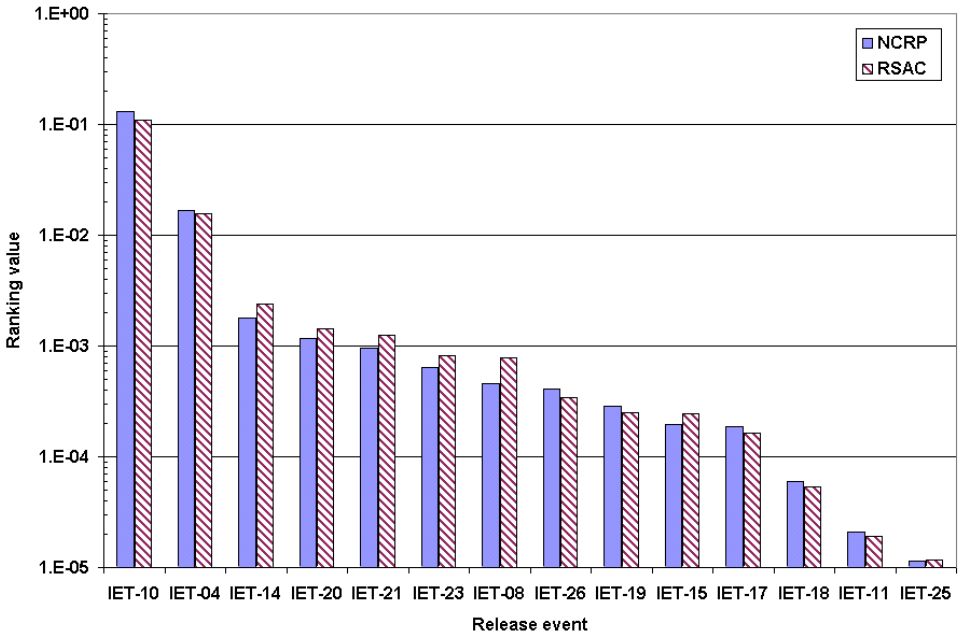


Figure 23. Offsite ranking values for Group 1 releases.

Table 19 shows the Group 1 ranking values by pathway. Table 20 shows the total Group 1 ranking values, which incorporate all relevant exposure pathways for each release. Only one significant digit is reported for the ranking values because uncertainties inherent in the process of estimating release, transport, and exposure limit the amount of precision that can be achieved. A

comparison of the ratios (NCRP value divided by RSAC value before rounding to a single significant digit) for each release and pathway suggests some general trends, although the differences between the two methodologies depend on the composition of the release (i.e., which radionuclides are present). These differences are discussed in more detail in Appendix B, but we provide some discussion here as it relates to the Group 1 releases.

Table 19. Group 1 Relative Ranking Values by Pathway

Test	Inhalation			Immersion			Ground irradiation			Ingestion		
	NCRP	RSAC	Ratio[a]	NCRP	RSAC	Ratio	NCRP	RSAC	Ratio	NCRP	RSAC	Ratio
IET-10	2.E-02	2.E-02	1.4	2.E-03	2.E-03	0.8	1.E-01	9.E-02	1.2	b		
IET-04	2.E-04	2.E-04	1.3	1.E-04	1.E-04	0.7	1.E-03	9.E-04	1.2	2.E-02	1.E-02	1.1
IET-14	2.E-05	1.E-05	1.4	3.E-06	5.E-06	0.7	8.E-05	7.E-05	1.2	2.E-03	2.E-03	0.7
IET-20	1.E-05	7.E-06	1.3	2.E-06	3.E-06	0.7	5.E-05	5.E-05	1.2	1.E-03	1.E-03	0.8
IET-21	1.E-05	8.E-06	1.3	2.E-06	3.E-06	0.7	5.E-05	4.E-05	1.2	9.E-04	1.E-03	0.8
IET-23	8.E-06	6.E-06	1.3	4.E-06	6.E-06	0.7	4.E-05	3.E-05	1.2	6.E-04	8.E-04	0.8
IET-08	1.E-05	1.E-05	1.0	5.E-05	7.E-05	0.7	3.E-04	4.E-05	8.3	1.E-04	7.E-04	0.1
IET-26	6.E-05	4.E-05	1.3	9.E-06	1.E-05	0.7	3.E-04	3.E-04	1.2	b		
IET-19	4.E-05	3.E-05	1.3	2.E-05	3.E-05	0.7	2.E-04	2.E-04	1.2	b		
IET-15	2.E-06	2.E-06	1.4	1.E-06	1.E-06	0.7	9.E-06	8.E-06	1.1	2.E-04	2.E-04	0.8
IET-17	3.E-05	2.E-05	1.4	1.E-05	2.E-05	0.7	1.E-04	1.E-04	1.2	b		
IET-18	1.E-05	7.E-06	1.4	7.E-06	1.E-05	0.7	4.E-05	4.E-05	1.2	b		
IET-11	3.E-06	2.E-06	1.4	3.E-06	4.E-06	0.7	2.E-05	1.E-05	1.2	b		
IET-25	2.E-06	1.E-06	1.4	3.E-06	5.E-06	0.7	6.E-06	5.E-06	1.2	b		

[a] Ratio equals NCRP value divided by RSAC value (before rounding to a single significant figure)
[b] Release occurred between November and April when the ingestion pathway was not considered

Table 20. Group 1 Total Relative Ranking Values

Test	Total ranking value		Ratio
	NCRP	RSAC	(NCRP/RSAC)
IET-10[a]	1.E-01	1.E-01	1.2
IET-04	2.E-02	2.E-02	1.1
IET-14	2.E-03	2.E-03	0.8
IET-20	1.E-03	1.E-03	0.8
IET-21	1.E-03	1.E-03	0.8
IET-23	6.E-04	8.E-04	0.8
IET-08	5.E-04	8.E-04	0.6
IET-26[a]	4.E-04	3.E-04	1.2
IET-19[a]	3.E-04	3.E-04	1.1
IET-15	2.E-04	2.E-04	0.8
IET-17[a]	2.E-04	2.E-04	1.1
IET-18[a]	6.E-05	5.E-05	1.1
IET-11[a]	2.E-05	2.E-05	1.1
IET-25[a]	1.E-05	1.E-05	1.0

[a] Ingestion pathway not included

For the inhalation pathway, the NCRP values are consistently about the same factor higher than the RSAC values. The small deviation for IET-8 relates to the fact that this release is dominated by ^{93}Mo instead of a mixture of various fission products. For the immersion pathway, the NCRP values are consistently about the same factor lower than the RSAC values. For the ground irradiation pathway, the NCRP values are consistently about the same factor higher than the RSAC values. The exception is again for IET-8, and this results from the difference between the two methodologies for ^{93}Mo. For the ingestion pathway, the NCRP values are generally slightly less than the RSAC values, although this depends again on the composition of the release, and IET-8 provides yet another deviation from the general trend owing again to differences between the two methodologies for ^{93}Mo. For those release events where the ingestion pathway was considered, it generally dominated the total ranking value. Where the ingestion pathway was not considered, the ground irradiation pathway generally dominated the total ranking value.

Onsite

Figure 24 provides a graphical representation of the onsite relative ranking values for each Group 1 release, sorted from highest to lowest according to the NCRP ranking values, which include the inhalation and immersion pathways only. The values obtained by the NCRP and RSAC methodologies are again quite comparable, and the two methodologies result in the same general relative ranking order, with some minor differences. The EPA methodology also results in the same general ranking order, again with some minor differences. Again, it is stressed that the relative ranked order suggested by each methodology is the desired result from these calculations, and the specific values calculated using the EPA methodology are not comparable to either the NCRP or RSAC methodologies.

As with the offsite ranking, the IET-10 release clearly appears as the highest-ranking release in Group 1, with IET-4 as the next highest-ranking release, regardless of the methodology that is used. However, comparing Figures 23 and 24 does reveal some differences in the overall offsite and onsite ranking orders. These differences are related to the fact that the composition of the release at the assumed exposure location varies as a function of the different onsite and offsite transit times and distances estimated for each release event, and the onsite exposure scenarios consider the inhalation and immersion pathways only. Onsite ranking values were not estimated for IET-11 and IET-17 because the trajectory of the release did not result in the plume passing over an onsite publicly accessible highway.

Based on this evaluation, IET-10 and IET-4 are the Group 1 release events that could be considered as a priority for any further evaluation that may be deemed necessary.

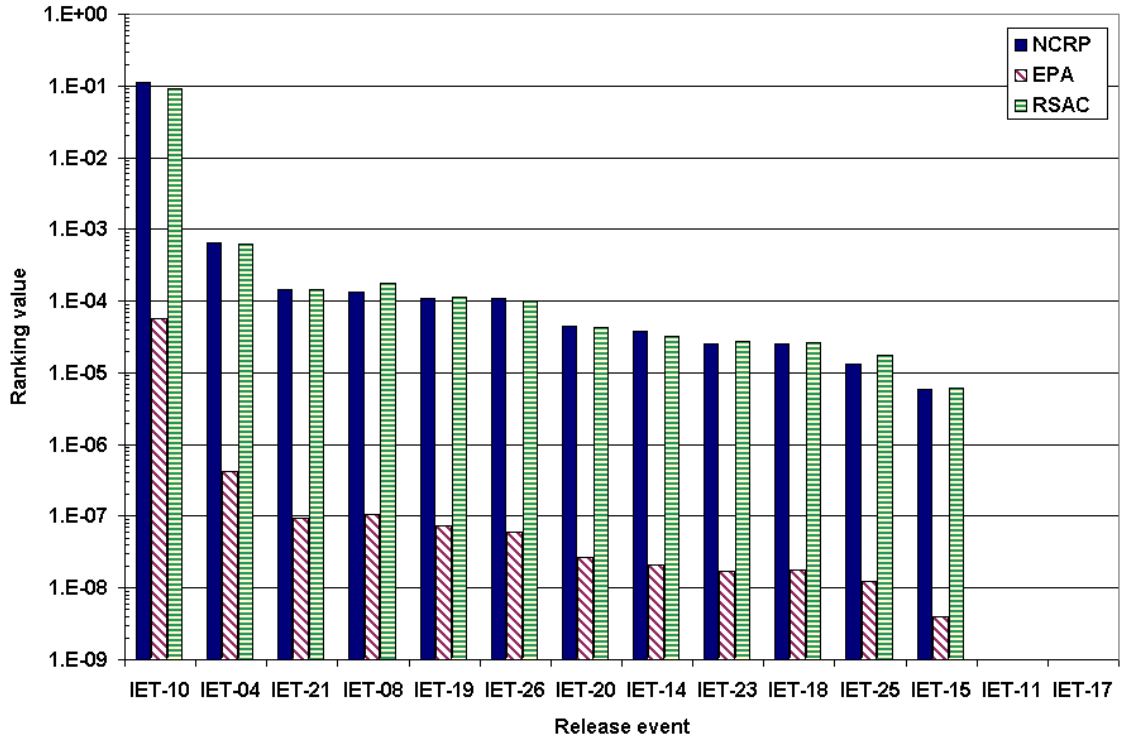


Figure 24. Onsite ranking values for Group 1 releases.

Group 2 Evaluation

As discussed previously, this group consists of shorter duration release events, requiring complete reconstruction of the composition of the release, and for which DOE (1991a) reported dispersion factors. Tables 21a and 21b show release-specific variables for each release event in this group. At the end of this section, relative ranking values are given for each release event.

Table 21a. Release-specific Variables for Group 2 Release Events

Release event	Dates of release: start	Dates of release: stop	Time of day	Hours modeled[a]	Season[b]	Ingestion[c]	Assumed mixing height (m)	Assumed wind speed (m s^{-1})	Assumed stability class
IET-3	2/11/56	2/24/56		35	winter	no	565	2.5[d]	D
IET-6	12/18/56			8	winter	no	565	2.0[e]	D
IET-12	5/6/58			4	spring		1405	3.0[e]	D
IET-13	11/18/58			6	autumn	no	940	2.8[e]	D
IET-16	10/9/59			33	summer		1580	7.6[e]	D
IET-22	8/25/60			2	summer		1580	6.7[e]	D
IET-24	10/26/60			1	autumn		940	9.0[e]	D
Borax	7/22/54		8:20 AM	8	summer		260	3.6[f]	E[g]
FEBT-A	3/20/57		2:19 PM	3	winter	no	730	5.8[f]	C
FEBT-B	3/20/57		6:47 PM	4	winter	no	730	2.7[f]	E
FPFRT-1	7/25/58		6:09 PM	15	summer		2900	4.6[f]	D
FPFRT-2	8/4/58		8:16 PM	9	summer		2900	4.0[f]	F
FPFRT-3	8/6/58		7:15 PM	27	summer		2900	2.7[f]	E
FPFRT-4	8/14/58		6:18 PM	28	summer		na	na	na
FPFRT-5	8/27/58		5:20 PM	17	summer		na	na	na
FPFRT-6	9/4/58		5:12 PM	53	summer		na	na	na
FPFRT-7	9/17/58		7:04 PM	40	summer		na	na	na
FPFRT-8	9/18/58		6:31 PM	23	summer		na	na	na
FPFRT-9	9/26/58		4:21 PM	56	autumn		1550	2.3[f]	D
FECF Filter Break	10/30/58		h	6	autumn		330	3.0[e]	D
1959 ICPP criticality	10/16/59		2:50 AM	3	autumn		330	6.3[f]	B
SL-1 accident	1/3/61		9:02 PM	16	winter	no	730	3.4[f]	F
1961 ICPP criticality	1/25/61		9:50 AM	5	winter	no	400	2.5[f]	C

Release event	Dates of release: start	Dates of release: stop	Time of day	Hours modeled[a]	Season[b]	Ingestion[c]	Assumed mixing height (m)	Assumed wind speed (m s^{-1})	Assumed stability class
SPERT-1	11/5/62		12:25 PM	2	autumn	no	1550	11.6[f]	D
SPERT-2	11/10/63		8:15 AM	8	autumn	no	330	6.5[f]	C
SPERT-3	4/14/64		1:14 PM	3	spring	no	2330	8.5[f]	C
SNAPTRAN-3	4/1/64		11:44 AM	1	spring	no	480	9.9[f]	C
SNAPTRAN-2	1/11/66		9:51 AM	6	winter	no	400	5.0[f]	D

[a] Source: DOE (1991a), Table B-2

[b] Spring: March 21 - June 20, Summer: June 21 - September 22, Autumn: September 23 - December 21, Winter: December 22 - March 20 (this seasonal division is approximate and some releases may overlap into another season by a few days)

[c] If release occurs between November and April, assumed no ingestion pathway contribution

[d] Wind speed and stability based on 5-year average at LOFT tower corresponding to specified time period of release

[e] Wind speed based on the distance and transit time to the Site boundary reported by DOE (1991a); stability selected to maximize onsite dispersion factor

[f] Wind speed and stability based on available documentation specific to release

[g] Stability selected to maximize onsite dispersion factor

[h] night of 10/29 and morning of 10/30

na – mixing height, wind speed, and stability estimates were not necessary because ratio of dispersion factors at different distances was not needed

Table 21b. Release-specific Variables for Group 2 Release Events

			Offsite				Onsite				
Release event	Release Location	Release height (m)	Location[a]	Downwind distance (km)	Transit time (minutes)	X/Q[b] (s m^{-3})	Location	Downwind distance (km)	Transit time (minutes)	X/Q ratio (offsite/onsite)	X/Q[c] (s m^{-3})
IET-3	TAN	46	Building	14	93	4.75E-07	HW 28	12	80	0.81	5.87E-07
IET-6	TAN	46	Atomic City	42	350	3.35E-10	HW 20	35	292	0.78	4.29E-10
IET-12	TAN	46	Reno Ranch	17	95	9.65E-10	HW 28	12	67	0.61	1.58E-09
IET-13	TAN	46	Howe	25	150	1.72E-07	HW 22	9	54	0.24	7.16E-07
IET-16	TAN	46	Cerro Grande	45	98	1.71E-08	HW 20	38	83	0.79	2.16E-08
IET-22	TAN	46	Monteview	16	40	4.68E-09	HW 28	10	25	0.52	9.00E-09
IET-24	TAN	46	Building	14	26	2.39E-08	HW 28	12	22	0.81	2.96E-08
Borax	ANL-W	0	Frenchman's Cabin	39	181	2.67E-08	HW 20	18	83	0.38	7.02E-08
FEBT-A	GRID III	0	Reno Ranch	47	135	3.64E-09	HW 33	24	69	0.56	6.49E-09
FEBT-B	GRID III	0	Birch Creek	56	346	5.69E-07	HW 22/33	21	130	0.30	1.90E-06
FPFRT-1	GRID III	0	Cellar	40	145	6.84E-09	HW 20	31	112	0.70	9.77E-09
FPFRT-2	GRID III	0	Blackfoot	62	258	1.10E-08	HW 20	13	54	0.16	6.86E-08
FPFRT-3	GRID III	0	Cellar	40	247	3.00E-07	HW 20	31	191	0.73	4.11E-07
FPFRT-4	GRID III	0	Mud Lake	45	147	3.51E-07	na	na	na	na	
FPFRT-5	GRID III	0	Mud Lake	45	95	1.67E-07	na	na	na	na	
FPFRT-6	GRID III	0	Roberts	67	167	2.44E-09	na	na	na	na	
FPFRT-7	GRID III	0	Mud Lake	45	132	1.56E-07	na	na	na	na	
FPFRT-8	GRID III	0	Mud Lake	45	163	2.10E-06	na	na	na	na	
FPFRT-9	GRID III	0	Atomic City	19	138	1.13E-07	HW 20	10	72	0.39	2.90E-07
FECF Filter Break	ICPP	0	Atomic City	17	94	1.74E-07	HW 20	9	50	0.40	4.35E-07
1959 ICPP criticality	ICPP	76	Frenchman's Cabin	21	56	7.45E-08	HW 20	6	16	0.33	2.26E-07

			Offsite				Onsite				
Release event	Release Location	Release height (m)	Location[a]	Downwind distance (km)	Transit time (minutes)	X/Q[b] (s m^{-3})	Location	Downwind distance (km)	Transit time (minutes)	X/Q ratio (offsite/onsite)	X/Q[c] (s m^{-3})
SL-1 accident	ARA	0	Atomic City	9	44	2.88E-07	HW 20	2	10	0.13	2.22E-06
1961 ICPP criticality	ICPP	76	Cerro Grande	14	93	3.49E-08	HW 20	6	40	0.44	7.94E-08
SPERT-1	SPERT	0	Building	44	63	3.02E-09	HW 33	36	52	0.76	3.98E-09
SPERT-2	SPERT	0	Cellar	35	90	1.27E-10	HW 20	20	51	0.61	2.08E-10
SPERT-3	SPERT	0	Mud Lake[d]	42	82	6.16E-09	na	na	na	0.72[e]	8.55E-09[f]
SNAPTRAN-3	TAN	46	Reno Ranch	17	29	1.32E-08	HW 28	12	20	0.73	1.80E-08
SNAPTRAN-2	TAN	46	Building	14	47	2.06E-08	HW 28	12	40	0.81	2.54E-08

[a] See Figure 21

[b] Source: Table B-2 of DOE (1991a)

[c] Estimated based on calculated ratio of offsite to onsite dispersion factors (X/Q)

[d] A location 2 miles south of Hamer was the offsite location assumed by DOE (1991a); because this was not one of the 16 offsite locations shown in Figure 21, we assumed Mud Lake

[e] Ratio was used to estimate the offsite dispersion factor for Mud Lake, based on the factor reported by DOE (1991a) for a location 2 miles south of Hamer

[f] Dispersion factor estimated for Mud Lake

na – onsite exposure scenario was not applicable because the trajectory of the release did not result in the plume passing over an onsite publicly accessible highway

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. This test series employed the HTRE-1 reactor assembly.

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 and was assumed by DOE (1991a) to occur through February 24, 1956. 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.

Thornton et al. (1962) reported activity releases of 2000 Ci during a 4-hour period and 1000 Ci during a 2-hour period during 100% nuclear operation, and releases of 100 Ci during a series of tests on the last day of operation. These data suggest a 500 curies per hour (Ci h^{-1}) release rate during 100% nuclear operation; however, a peak release rate of approximately 30 Ci h^{-1} particulate activity is also suggested for IET-3 in Figure 5.8 of Thornton et al. (1962). Conservatively assuming a 500 Ci h^{-1} rate of release during the entire 40 hours of operation above 200 kW results in a total particulate release of 20,000 Ci. We used RSAC to calculate an estimated fission product inventory, based on the operational history provided by Thornton et al. (1962). We adjusted the release fractions until our reconstructed operation resulted in a total 10-minute decayed particulate release of approximately 21,085 Ci. The resulting release fractions were 0.0032, 0.016, and 0.032 for solids, halogens, and noble gases, respectively.

Thornton et al. (1962) reported that the total uranium (93.4% enriched) in the HTRE-1 core was 90 pounds (40,800 grams). We assumed the uranium in the core was released in the same fraction used for the solids and estimated production and consequent release of 1060 Ci ^{41}Ar (2.8 Ci MW-h^{-1} times a total power generation of 378.5 MW-h).

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). The first indication that some amount of damage had occurred within the A-3 core was detected on the night of December 18, 1956. DOE (1991a) assumed that all releases to the atmosphere occurred on December 18, 1956. 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 to determine the cause of fuel cartridge damage and to prepare for operation of the A3 reactor (Thornton et al. 1962).

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 approximately 6350 Ci. We used RSAC to calculate an estimated fission product inventory, based on the operational history provided by Thornton et al. (1962). We adjusted the release fractions until our reconstructed operation resulted in a total 10-minute decayed particulate release of approximately 6817 Ci. The resulting release fractions were 0.0015, 0.0075, and 0.015 for solids, halogens, and noble gases, respectively.

Thornton et al. (1962) reported that the total uranium (93.4% enriched) in the HTRE-1 core was 90 pounds (40,800 grams). We assumed the uranium in the core was released in the same fraction used for the solids and estimated production and consequent release of 8584 Ci ^{41}Ar (2.8 Ci MW-h^{-1} times a total power generation of 3065.6 MW-h).

IET-12

The IET-12 test series, also referred to as Operation Burnout One Tube (BOOT), was carried out between April 21 and May 6, 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. Baker et al. (1959) reported 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. 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.

Devens et al. (1958) reported 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. We used RSAC to calculate an estimated fission product inventory, based on the operational history provided by Devens et al. (1958) and further refined by DOE (1991a). We adjusted the release fractions until our reconstructed operation resulted in a total 10-minute decayed release of approximately 21,600 Ci. The resulting release fractions were 0.0048, 0.024, 0.048 for solids, halogens, and noble gases, respectively.

Flagella (1962) reported that the total uranium (93.4% enriched) in the HTRE-2 core was 95 pounds (43,100 grams). We assumed 843 grams of uranium was released based on the information reported by Baker et al. (1959) and estimated production and consequent release of 79 Ci ^{41}Ar (2.8 Ci MW-h^{-1} times a total power generation of 28 MW-h).

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.

Wilks (1959) reported a gross fission product release of 400 Ci. We used RSAC to calculate an estimated fission product inventory, based on the operational history provided by Devens (no date). We adjusted the release fractions until our reconstructed operation resulted in a total 10-minute decayed release of approximately 416 Ci. The resulting release fractions were 0.0003, 0.0015, 0.003 for solids, halogens, and noble gases, respectively.

GE (1959) reported that the total uranium (93.4% enriched) in the HTRE-3 core was 425 pounds (193,000 grams). We assumed the uranium in the core was released in the same fraction used for solids and estimated production and consequent release of 1 Ci ^{41}Ar (2.8 Ci MW-h^{-1} times a total power generation of 0.3 MW-h).

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. Because of the relatively low power generation during this test series, consequent releases of radioactivity to the atmosphere were also relatively small. DOE (1991a) assumed that all releases to the atmosphere occurred on October 9, 1959.

Showalter et al. (1959) reported 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 14 Ci. We used RSAC to calculate an estimated fission product inventory, based on a simplified treatment of the operational history provided by Showalter et al. (1959). Miller et al. (1960) reported 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. We adjusted the release fractions until our reconstructed operation resulted in a total 10-minute decayed release of approximately 14 Ci. The resulting release fractions were 0.00000046, 0.0000023, 0.0000046 for solids, halogens, and noble gases, respectively.

GE (1959) reported that the total uranium (93.4% enriched) in the HTRE-3 core was 425 pounds (193,000 grams). We assumed the uranium in the core was released in the same fraction used for solids and estimated production and consequent release of 266 Ci ^{41}Ar (2.8 Ci MW-h^{-1} times a total power generation of 95.0 MW-h).

IET-22

The IET-22 test series, or Limited Melt Experiment No. 1 (LIME-I), was conducted between August 12 and 25, 1960. The reactor was operated on August 22 and 23 in preparation for the LIME experiment on August 25, when the majority of atmospheric releases were assumed by DOE (1991a) to occur. The test series employed the HTRE-2 reactor assembly and involved evaluation of the L2E-4 insert cartridge, which consisted of uncoated fueled and unfueled BeO tubes. The center tube was an instrumentation tube, around which were two rows of fueled tubes that were blocked from any air flow. 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. Post-operational examination showed that melting temperature had indeed been

reached in the center of the cartridge but that the damage had almost entirely been contained in the blocked fuel region (Pincock 1960d).

Pincock (1960d) reported a gross fission product release of 3250 Ci of 10-minute decayed activity. We used RSAC to calculate an estimated fission product inventory, based on the operational history provided by Pincock (1960d). We adjusted the release fractions until our reconstructed operation resulted in a total 10-minute decayed release of approximately 3250 Ci. The resulting release fractions were 0.00048, 0.0024, 0.0048 for solids, halogens, and noble gases, respectively.

Flagella (1962) reported that the total uranium (93.4% enriched) in the HTRE-2 core was 95 pounds (43,100 grams). We assumed 843 grams of uranium was released based on the information reported by Baker et al. (1959) and estimated production and consequent release of 27 Ci ^{41}Ar (2.8 Ci MW-h^{-1} times a total power generation of 9.6 MW-h).

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 reactor was operated on October 21 and 24 in preparation for the LIME-II experiment on October 26, when the majority of atmospheric releases were assumed by DOE (1991a) to occur. The test series employed the HTRE-2 reactor assembly and 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. Post-operational examination showed that stages 6 through 11 were fused end to end in the fueled region and all tubes in the first three rings of stages 8, 9, and 10 were fused. All non-fueled tubes in stages 1 through 10 were whole and apparently undamaged. All fuel tubes that did not fuse were whole and apparently undamaged, and all fused tubes had numerous transverse and cross-sectional cracks. The center tubes of the fueled region had a heavy deposit of white crystals (Pincock 1960e; Baker 1961).

Pincock (1960e) reported a gross fission product release of 1880 Ci of 10-minute decayed activity. We used RSAC to calculate an estimated fission product inventory, based on the operational history provided by Pincock (1960e). We adjusted the release fractions until our reconstructed operation resulted in a total 10-minute decayed release of approximately 1950 Ci. The resulting release fractions were 0.00048, 0.0024, 0.0048 for solids, halogens, and noble gases, respectively.

Flagella (1962) reported that the total uranium (93.4% enriched) in the HTRE-2 core was 95 pounds (43,100 grams). We assumed 843 grams of uranium was released based on the information reported by Baker et al. (1959) and estimated production and consequent release of 71 Ci ^{41}Ar (2.8 Ci MW-h^{-1} times a total power generation of 25.4 MW-h).

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 ft. Monitoring teams dispatched shortly after the excursion determined the trajectory of the cloud to have been roughly in a southwesterly 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 (3.6 m s^{-1}) at ground level and 20 mph at the 250-foot level.

Based on the reactor operating parameters provided by Dietrich (1954) and DOE (1991a), we used RSAC 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.

Dietrich (1954) reported that "The actual energy of the excursion proved to be about 135 megawatts, and, instead of the melting of a few fuel plates, the melting of the major fraction of the entire core was accomplished." Based on calculations noted to be "very rough" and ground surveys "necessarily far from precise", the report noted that "…the survey gives no indication that any large fraction of the fission products left the vicinity of the reactor." Because of the uncertainties associated with this information, we assumed release fractions of 1.0 for noble gases, 0.5 for halogens, and 0.1 for solids and uranium in the fuel. Dietrich (1954) reported 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. Generation of ^{41}Ar would be expected to be negligible because the reactor was both water-cooled and moderated; therefore, this radionuclide was not incorporated into our reconstructed release.

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 ft 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., decreasing temperature with increasing elevation, which allows for greater upward dispersion), and, therefore, optimum diffusion conditions (Brodsky and Beard 1960).

Brodsky and Beard (1960) reported 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 0.55 MW for 120 hours followed by a decay of 70 days for our inventory calculation with RSAC. These operating parameters lead to a fission product inventory of 5150 Ci. Uranium activity was 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) reported only that "probably less than a few tenths of a per cent" of the fuel element inventory was released. Because this does not provide us with specific release fractions for individual radionuclides, we assumed release fractions of 0.1, 0.5, and 1.0 for solids (including uranium in the fuel element), halogens, and noble gases, respectively.

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 ft 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) reported 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 used RSAC to arrive at a calculated inventory of 10,200 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. 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 assumed release fractions of 0.1, 0.5, and 1.0 for solids (including uranium in the fuel element), halogens, and noble gases, respectively.

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 at test GRID III 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 22.

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). Convair (1959) provided the operational histories and estimated fission product inventories for each of the elements. To simulate a potential accident, they used an induction type furnace to rapidly heat the elements to the melting point in approximately 2 minutes, and

maintained 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 22. 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 RSAC 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 23 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. 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. The reason for these discrepancies was not apparent. For our ranking calculations, we used the RSAC-generated inventory. Uranium releases were estimated by assuming a total 90% enriched uranium content of 222 g for each element, and the same release fraction assumed for solids for each test.

Table 23. Comparison of Calculated and Reported Radionuclide Inventories for the Fuel Element Used During FPFRT-5

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

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). A description of these methods is provided here.

The first method described by Convair (1959) that was used to estimate release fractions 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 24 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) provided 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 24. 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, which 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 25 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 25. 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).

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) reported 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. Because the majority of the activity released from the FECF was associated with very large particles and rapidly deposited on the roof of and ground near the FECF, the release fraction for all fission products was adjusted to result in a total release of approximately 110 Ci to correspond with the reported amount of activity deposited outside the perimeter fence (a final release fraction of 0.02 was assumed for all fission products). We assumed the same uranium release values estimated by DOE (1991a).

Table 25. 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.

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. Based on available evidence, it was determined that the critical condition resulted from the accidental transfer of a concentrated uranyl nitrate solution from geometrically safe storage banks in a process cell into a waste collection tank through a line normally, used to transfer decontaminating solutions to waste. Siphon action initiated by air sparging was indicated to be the most likely mechanism by which the transfer took place (Ginkel et al. 1960). 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 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 as assumed by RSAC. Lewis (1960, Exhibit D to Ginkel et al. 1960) indicated that the reaction occurred over a period of several minutes, so we assumed a reactor power level of 2.14 MW for a period of 10 minutes (600 seconds).

Limited information was available regarding the fraction of the inventory that may have been released. However, Anonymous (no date) noted a total release of 350,000 Ci of primarily short-lived gaseous fission products, and Hayden (1959) estimated the total ^{131}I release to be 3.73 Ci, based on analyses of the stack-gas-monitor scrubber solution. Hayden (1959) reported an

estimated release for ^{132}I; 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, this information was not used to reconstruct the release. In addition to the decay occurring during transit from the point of release to the point of exposure, decay was incorporated to account for the time to travel from the collection stack and exit the stack. The time allowed for this decay was adjusted iteratively to arrive at a total activity at the time of release of approximately 350,000 Ci (the necessary time for decay was approximately 180 seconds). Because this assumed reactor operation produces a direct yeild much less than the amount of ^{131}I reported to be released, calculations were made at various decay times to determine the maximum amount of each isotope of iodine that could occur following the criticality. The reported amount of ^{131}I (3.73 Ci) was divided by this maximum calculated value to derive a release fraction for halogens (0.143). Release fractions of 1.0 and 0.1 were assumed for noble gases and solids, respectively. These calculations result in an estimated 3-minute decayed release of ^{132}I of approximately 14 Ci versus the 9,780 Ci release reported by Hayden (1959).

Stationary Low-Power Reactor No. 1 Accident

The Stationary Low-Power Reactor No. 1 (SL-1) was the smallest known power reactor when it began critical operations in August 1958 at the Auxiliary Reactor Area (ARA). 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. The SL-1 was shutdown on December 23, 1960, for the installation of cobalt wires to be used in mapping the neutron flux of the reactor. On the evening of January 3, 1961, all wires had been installed and a three man operating crew was preparing the reactor for startup. 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 RSAC 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 information 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) reported 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 93% 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. No ^{41}Ar generation was assumed to occur.

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 criticality was determined to have resulted from the accidental lifting of a solution of enriched uranyl nitrate from the lower, geometrically safe section of the evaporatory into the upper, 24-inch diameter, critically unsafe, vapor-disengagement section. The most likely cause of the lift appeared to have been a sudden burst of air inadvertently introduced in the bottom of the evaporator (Paulus et al. 1961).

The period following the nuclear incident was characterized by light northerly winds and a temperature lapse up to 500 ft 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, and the energy release apparently occurred as a single burst (Paulus et al. 1961). A fission product inventory was calculated using the RSAC 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. Because the reaction was reported to occur as a single burst, we assumed a reactor power level of 20 MW for 1 second.

Limited information was available regarding the fraction of the inventory that may have been released. However, Anonymous (no date) noted a total release of 5,200 Ci of primarily short-lived gaseous fission products. No 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. In addition to the decay occurring during transit from the point of release to the point of exposure, decay was incorporated to account for the time to travel from the collection stack and exit the stack. The time allowed for this decay was adjusted iteratively to arrive at a total activity at the time of release of approximately 5,200 Ci (the necessary time for decay was approximately 410 seconds).

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 26.

Table 26. 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)[b]	0.25	0.20	0.20

[a]Source: Bunch (1965)

[b]A stability parameter of 0.25 corresponds to neutral atmospheric stability; although this was not noted by the authors, we assumed that the lower values of 0.20 for Tests 2 and 3 correspond to somewhat less stable (i.e., lapse) conditions, based on the weather requirements for the test.

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) reported that less than 0.01% of the iodines was released to the atmosphere. Based on this information, we conservatively 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. Because of the location of the reactor under 11 or 12 ft of water (DOE 1991b), production of ^{41}Ar was assumed to be insignificant.

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, as was production of ^{41}Ar. Based on this information, we assumed release fractions of 0.0002 for noble gases, 0.0001 for halogens, and 0 for solids.

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, as was production of ^{41}Ar. Based on this information, we assumed release fractions of 0.0006 for noble gases, 0.0001 for halogens, and 0 for solids.

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).

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 27.

Table 27. 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[b]	209°	17.7
Station Monteview	197°	18.6
Tetroon[c] (released after test)	209°	27.9

[a] Source: Cordes et al. (1965)
[b] Along Idaho Highway 28
[c] 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) reported that "the upper limit of noble gas release…is four percent." We assumed a release fraction of 0.04 for noble gases, a conservative (because none were detected) release fraction of 0.02 for halogens, and a release fraction of 0.0 for particulate radionuclides. Because the reactor was surrounded by water, no ^{41}Ar generation was assumed.

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) reported 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 (Cordes et al. 1967). 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. The reactor operation was assumed to generate 4681 Ci of ^{41}Ar. This estimate was derived using the methodology described by DOE (1991b), which was based on an assumed volume of air surrounding the reactor, an estimate of the amount of argon gas in that volume of air, and an estimate of the number of atoms of argon activated by neutrons escaping the reactor core.

Group 2 Ranking Results

The following sections discuss the results of both the offsite and onsite ranking values for the Group 2 releases.

Offsite

Figure 25 provides a graphical representation of the offsite relative ranking values for each Group 2 release, sorted from highest to lowest according to the NCRP total ranking values (Table 29). The values obtained by the NCRP and RSAC methodologies are quite comparable, and both methodologies result in the same general relative ranking. Although there are some minor differences in ordering between the two methodologies, six releases stand out as the highest-

ranking releases in Group 2. These highest-ranking releases include FEBT-B, FPFRT-8, FPFRT-4, IET-3, FECF filter break, and FPFRT-3.

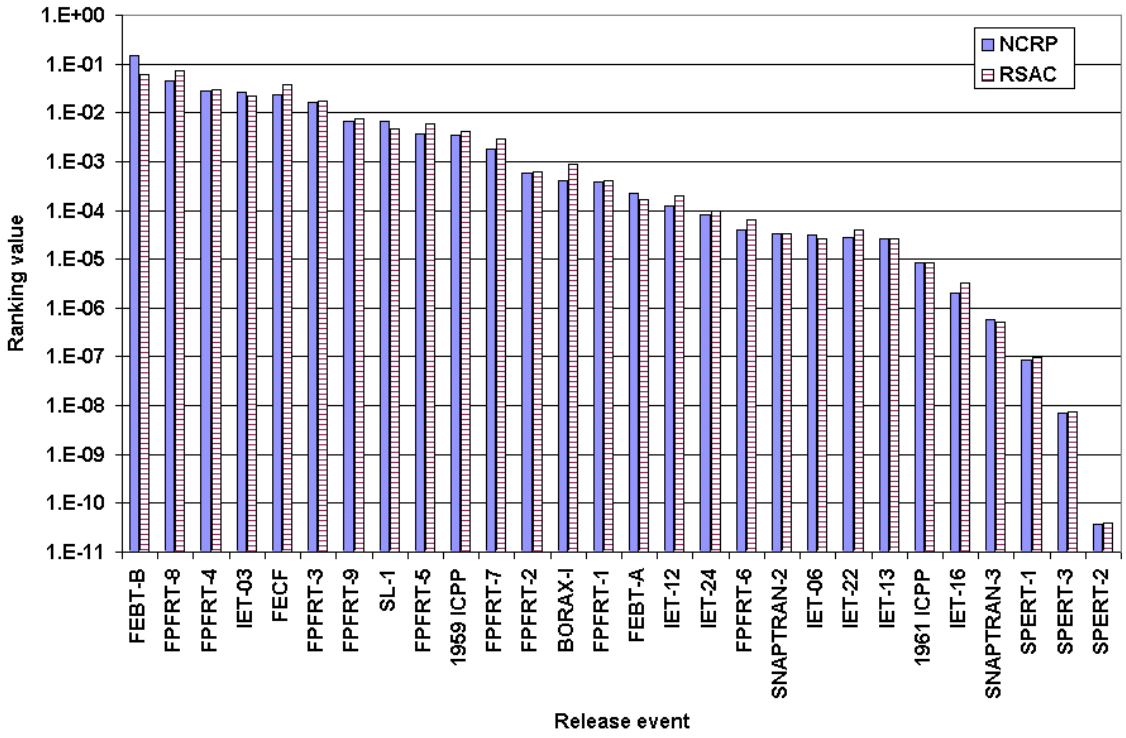


Figure 25. Offsite ranking values for Group 2 releases.

Table 28 shows the Group 2 ranking values by pathway. Table 29 shows the total Group 2 ranking values, which incorporate all relevant exposure pathways for each release. Again, a single significant digit is reported, and trends in the ratios are generally similar to those discussed for the Group 1 releases. Notably, though, the NCRP methodology results in significantly higher values for the ground irradiation pathway when the release is comprised primarily of longer-lived radionuclides (e.g., FPFRT-1, -2, -3, -4, and -9). This result was not unexpected, as the NCRP screening factors incorporate an assumed 30-year buildup period for radionuclides in the soil. As with the Group 1 releases, the ingestion pathway generally dominates the total ranking value for those releases where it is considered; however, for releases involving longer-lived radionuclides, the ground irradiation pathway becomes a significant contributor as well. As noted previously, differences between the two methodologies are discussed in more detail in Appendix B.

Table 28. Group 2 Relative Ranking Values by Pathway

Test	Inhalation			Immersion			Ground irradiation			Ingestion		
	NCRP	RSAC	Ratio[a]	NCRP	RSAC	Ratio	NCRP	RSAC	Ratio	NCRP	RSAC	Ratio
FEBT-B	2.E-02	2.E-02	1.0	2.E-05	3.E-05	0.8	1.E-01	4.E-02	2.9	b		
FPFRT-8	6.E-04	6.E-04	1.0	3.E-06	4.E-06	0.8	1.E-02	2.E-03	7.7	3.E-02	7.E-02	0.4
FPFRT-4	6.E-05	6.E-05	1.0	9.E-08	1.E-07	0.8	1.E-02	6.E-04	20.4	2.E-02	3.E-02	0.5
IET-03	4.E-03	3.E-03	1.4	2.E-03	2.E-03	0.7	2.E-02	2.E-02	1.2	b		
FECF	7.E-04	7.E-04	1.0	5.E-07	7.E-07	0.8	6.E-03	1.E-03	4.9	2.E-02	4.E-02	0.5
FPFRT-3	3.E-05	3.E-05	1.0	5.E-08	7.E-08	0.8	7.E-03	4.E-04	20.5	9.E-03	2.E-02	0.5
FPFRT-9	1.E-05	1.E-05	1.0	2.E-08	3.E-08	0.8	3.E-03	1.E-04	20.5	4.E-03	7.E-03	0.5
SL-1	8.E-04	7.E-04	1.2	2.E-04	3.E-04	0.8	6.E-03	4.E-03	1.6	b		
FPFRT-5	4.E-05	3.E-05	1.1	4.E-07	5.E-07	0.8	7.E-04	2.E-04	4.4	3.E-03	6.E-03	0.5
1959 ICPP[c]	1.E-04	1.E-04	1.3	4.E-04	7.E-04	0.7	6.E-04	5.E-04	1.2	2.E-03	3.E-03	0.8
FPFRT-7	2.E-05	2.E-05	1.0	1.E-07	2.E-07	0.8	6.E-04	8.E-05	7.6	1.E-03	3.E-03	0.4
FPFRT-2	1.E-06	1.E-06	1.0	2.E-09	2.E-09	0.8	3.E-04	1.E-05	20.5	3.E-04	6.E-04	0.5
BORAX-I	4.E-05	4.E-05	1.0	4.E-06	5.E-06	0.7	9.E-05	6.E-05	1.6	3.E-04	8.E-04	0.3
FPFRT-1	7.E-07	7.E-07	1.0	1.E-09	2.E-09	0.8	2.E-04	8.E-06	20.4	2.E-04	4.E-04	0.5
FEBT-A	2.E-05	2.E-05	1.0	1.E-07	1.E-07	0.8	2.E-04	2.E-04	1.3	b		
IET-12	3.E-06	3.E-06	1.1	7.E-07	1.E-06	0.7	6.E-06	5.E-06	1.1	1.E-04	2.E-04	0.6
IET-24	3.E-06	2.E-06	1.4	6.E-06	9.E-06	0.7	1.E-05	1.E-05	1.2	6.E-05	8.E-05	0.8
FPFRT-6	4.E-07	4.E-07	1.1	3.E-09	5.E-09	0.8	1.E-05	2.E-06	5.5	3.E-05	6.E-05	0.5
SNAPTRAN-2	1.E-05	1.E-05	1.1	8.E-06	1.E-05	0.7	1.E-05	1.E-05	1.2	b		
IET-06	4.E-06	3.E-06	1.2	1.E-07	2.E-07	0.7	3.E-05	2.E-05	1.2	b		
IET-22	7.E-07	5.E-07	1.3	1.E-06	1.E-06	0.7	2.E-06	2.E-06	1.2	2.E-05	4.E-05	0.7
IET-13	2.E-05	2.E-05	1.0	6.E-07	9.E-07	0.7	3.E-06	3.E-06	1.2	b		
1961 ICPP[c]	1.E-06	1.E-06	1.2	2.E-06	3.E-06	0.7	6.E-06	5.E-06	1.1	b		
IET-16	4.E-07	3.E-07	1.0	4.E-07	7.E-07	0.7	1.E-07	8.E-08	1.2	1.E-06	2.E-06	0.5
SNAPTRAN-3	5.E-08	3.E-08	1.5	2.E-07	3.E-07	0.8	3.E-07	2.E-07	1.4	b		
SPERT-1	8.E-09	8.E-09	1.0	3.E-08	4.E-08	0.7	5.E-08	5.E-08	1.1	b		
SPERT-3	8.E-10	7.E-10	1.0	2.E-09	3.E-09	0.7	5.E-09	4.E-09	1.1	b		
SPERT-2	5.E-12	4.E-12	1.1	9.E-12	1.E-11	0.7	2.E-11	2.E-11	1.1	b		

[a] Ratio equals NCRP value divided by RSAC value (before rounding to a single significant figure)

[b] Release occurred between November and April when the ingestion pathway was not considered

[c] Criticality event

Table 29. Group 2 Total Relative Ranking Values

Test	Total ranking value		Ratio
	NCRP	RSAC	(NCRP/RSAC)
FEBT-B[a]	1.E-01	6.E-02	2.4
FPFRT-8	5.E-02	7.E-02	0.6
FPFRT-4	3.E-02	3.E-02	0.9
IET-03[a]	3.E-02	2.E-02	1.1
FECF	2.E-02	4.E-02	0.6
FPFRT-3	2.E-02	2.E-02	0.9
FPFRT-9	7.E-03	8.E-03	0.9
SL-1[a]	7.E-03	5.E-03	1.5
FPFRT-5	4.E-03	6.E-03	0.6
1959 ICPP	4.E-03	4.E-03	0.9
FPFRT-7	2.E-03	3.E-03	0.6
FPFRT-2	6.E-04	6.E-04	0.9
BORAX-I	4.E-04	9.E-04	0.5
FPFRT-1	4.E-04	4.E-04	0.9
FEBT-A[a]	2.E-04	2.E-04	1.3
IET-12	1.E-04	2.E-04	0.6
IET-24	8.E-05	1.E-04	0.8
FPFRT-6	4.E-05	6.E-05	0.6
SNAPTRAN-2[a]	3.E-05	3.E-05	1.0
IET-06[a]	3.E-05	3.E-05	1.2
IET-22	3.E-05	4.E-05	0.7
IET-13[a]	3.E-05	3.E-05	1.0
1961 ICPP[a]	9.E-06	9.E-06	1.0
IET-16	2.E-06	3.E-06	0.6
SNAPTRAN-3[a]	6.E-07	5.E-07	1.1
SPERT-1[a]	9.E-08	9.E-08	0.9
SPERT-3[a]	7.E-09	8.E-09	0.9
SPERT-2[a]	4.E-11	4.E-11	1.0

[a] Ingestion pathway not included

Onsite

Figure 26 provides a graphical represenation of the onsite relative ranking values for each Group 2 release, sorted from highest to lowest according to the NCRP ranking values, which include the inhalation and immersion pathways only. As with the onsite values for the Group 1 releases, the NCRP and RSAC values are quite comparable. Both of these methodologies and the EPA methodology result in the same general relative ranking order, with some minor differences. Again, it is stressed that the relative ranked order suggested by each methodology is the desired result from these calculations, and the specific values calculated using the EPA methodology are not comparable to either the NCRP or RSAC methodologies.

As with the offsite ranking for Group 2, several releases stand out as the highest-ranking releases in Group 2. The FEBT-B, IET-3, and FECF filter break are among the highest-ranking releases with regard to onsite exposure that also ranked highly in the offsite exposure evaluation. The SL-1 and 1959 ICPP releases are also among the highest-ranking releases with regard to onsite exposure. As with the Group 1 releases, comparing Figures 25 and 26 does reveal some differences in the overall offsite and onsite ranking orders. These differences are related to the fact that the composition of the release at the assumed exposure location varies as a function of the different onsite and offsite transit times and distances estimated for each release event, and the onsite exposure scenarios consider the inhalation and immersion pathways only. Onsite ranking values were not estimated for FPFRT-4, -5, -6, -7, and -8 and SPERT-3 because the trajectory of the release did not result in the plume passing over an onsite publicly accessible highway.

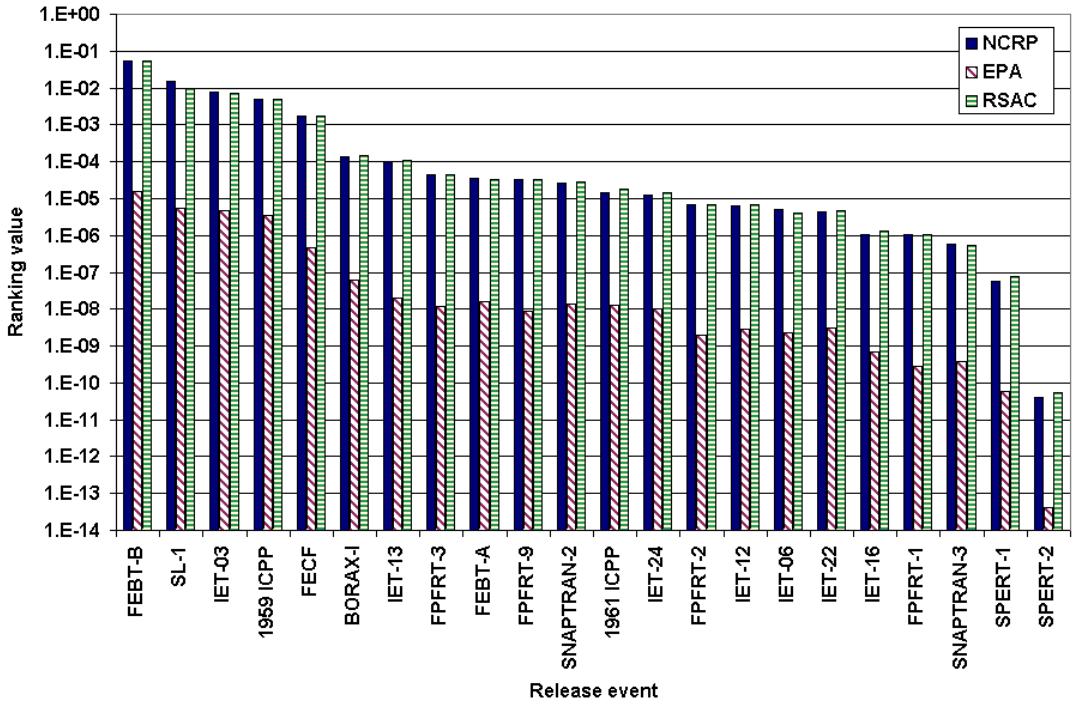


Figure 26. Onsite ranking values for Group 2 releases.

Based on this evaluation, the FEBT-B, FPFRT-8, FPFRT-4, IET-3, FECF filter break, and FPFRT-3 releases are the Group 2 releases that could be considered as a priority for any further evaluation that may be deemed necessary. In addition, the SL-1 and 1959 ICPP releases may also deserve consideration if onsite exposure scenarios are determined to be important as part of any further evaluations.

Group 3 Evaluation[8]

As discussed previously, Group 3 consists of release events for which known quantities of radionuclides were released (i.e., reconstruction of the release was not required) and for which DOE (1991a) reported dispersion factors. With the exception of the NRF S1W Engineering Test release, these are shorter duration release events. Tables 30a and 30b show release-specific variables for each release event in this group. At the end of this section, relative ranking values are given for each Group 3 release event.

[8] Based on the close agreement between the NCRP and RSAC methodologies for the routine and Group 1 and 2 releases and the detailed comparison of the two methodologies presented in Appendix B, use of the NCRP screening factors for the remaining Group 3, 4, and 5 release events is considered defensible and appropriate. Therefore, ranking values for the remaining releases are calculated using the NCRP methodology only.

Table 30a. Release-specific Variables for Group 3 Release Events

Release event	Dates of release start	Dates of release stop	Time of day	Hours modeled[a]	Season[b]	Ingestion[c]	Assumed mixing height (m)	Assumed wind speed (m/s)	Assumed stability class[d]
NRF S1W Engineering Test	6/18/55	6/30/55		350	spring/summer		1580	4.7[e]	D
CERT #1	5/27/63		3:00 PM	40	spring		2330	5.2[f]	D
CERT #2	9/2/64		1:44 PM	2	summer		na	na	na
CERT #3	12/11/64		1:54 PM	13	autumn	no	na	na	na
CERT #4	5/27/65		na	12	spring		na	na	na
CERT #5	6/10/65		5:15 AM	3	spring		480	2.7[f]	D
CERT #6	9/14/65		2:00 PM	2	summer		2900	15[f]	C
CERT #7	11/22/65		2:10 PM	26	autumn	no	na	na	na
CERT #10	6/14/66		10:40 AM	4	spring		na	na	na
CERT #11	7/21/66		1:30 PM	3	summer		2900	5[f]	C
CERT #20	7/6/67		2:01 PM	2	summer		2900	10.6[g]	C
CERT #22	9/22/67		2:30 PM	2	summer		2900	8.9[g]	C

[a] Source: DOE (1991a), Table B-2

[b] Spring: March 21 - June 20, Summer: June 21 - September 22, Autumn: September 23 - December 21, Winter: December 22 - March 20

[c] If release occurs between November and April, assumed no ingestion pathway contribution

na – mixing height, wind speed, and stability estimates were not necessary because because onsite exposure scenario was not applicable

[d] Stability class selected to maximize onsite dispersion factor

[e] Wind speed and stability based on 5-year average at GRID III tower corresponding to specified time period of release

[f] Wind speed based on available documentation specific to release

[g] Wind speed based on the distance and transit time to the Site boundary reported by DOE (1991a)

Table 30b. Release-specific Variables for Group 3 Release Events

			Offsite				Onsite				
Release event	Release Location	Release height (m)	Location[a]	Downwind distance (km)	Transit time (minutes)	X/Q[b] (s/m3)	Location	Downwind distance (km)	Transit time (minutes)	X/Q ratio (offsite/onsite)	X/Q[c] (s/m3)
NRF S1W Engineering Test	NRF N. of Atomic	0	Building	40	103	5.04E-08	HW 33	29	103	0.64	7.88E-08
CERT #1	City	0	Idaho Falls	63	na	1.85E-08	na	na	na	na	
CERT #2	EFS	0	Mud Lake	44	na	1.54E-08	na	na	na	na	
CERT #3	EFS	0	Roberts	66	na	7.06E-08	na	na	na	na	
CERT #4	ARA	0	Mud Lake	41	na	4.21E-10	na	na	na	na	
CERT #5	ARA	0	Atomic City	10	na	1.58E-08	HW 20	4	na	0.25	6.31E-08
CERT #6	ICPP	76	Building	46	na	2.37E-09	HW 33	37	na	0.70	3.38E-09
CERT #7	EFS	0	Howe	20	na	2.28E-08	na	na	na	na	
CERT #10	GRID III	0	Roberts	67	na	9.47E-10	na	na	na	na	
CERT #11	EFS	0	Building	44	na	3.23E-09	HW 33	35	na	0.68	4.75E-09
CERT #20	EFS	0	Monteview	49	na	2.51E-09	HW 33	33	na	0.55	4.57E-09
CERT #22	EFS	0	Reno Ranch	46	na	1.24E-08	HW 22/33	23	na	0.30	4.14E-08

[a] See Figure 21

[b] Source: Table B-2 of DOE (1991a)

[c] Estimated based on calculated ratio of offsite to onsite dispersion factors (X/Q)

na – onsite exposure scenario was not applicable because the trajectory of the release did not result in the plume passing over an onsite publicly accessible highway

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). Because this release was comprised of several short-lived radionuclides, we decayed the effluent data at the point of release according to the times estimated for transit to the onsite and offsite exposure locations.

Selected CERT Releases

The remaining Group 3 releases consist only of the Controlled Environmental Radioiodine Test (CERT) releases that were specifically evaluated as episodic releases by DOE (1991a); therefore, an estimated dispersion factor for each release was available. These releases, along with other release tests, are discussed in more detail in the Group 4 evaluation section.

Group 3 Ranking Results

Figure 27 provides a graphical representation of the offsite and onsite ranking values for each Group 3 release, sorted from highest to lowest according to the offsite ranking values (Table 31). As with the Group 1 and 2 releases, there are clearly differences in the overall offsite and onsite ranking orders. These differences are related to the different onsite and offsite distances estimated for each release event, and the onsite exposure scenarios consider the inhalation and immersion pathways only. Onsite ranking values were not estimated for CERT #1, 2, 3, 4, 7, and 10 because the trajectory of the release did not result in the plume passing over an onsite publicly accessible highway. For the Group 3 releases, the NRF S1W Engineering Test release is clearly the highest-ranking release with regard to both onsite and offsite exposure scenarios.

Although the ranking values for the Group 3 releases are not directly comparable to ranking values established for releases in other groups because of differences in the assumptions required to evaluate the releases in each group, some general conclusions can be drawn based on the magnitude of the ranking values. Because the highest ranking values for the Group 3 releases are several orders of magnitude lower than the highest ranking values for either the Group 1 or Group 2 releases, it is very likely that their overall relative importance in terms of health impacts to exposed individuals is also much less.

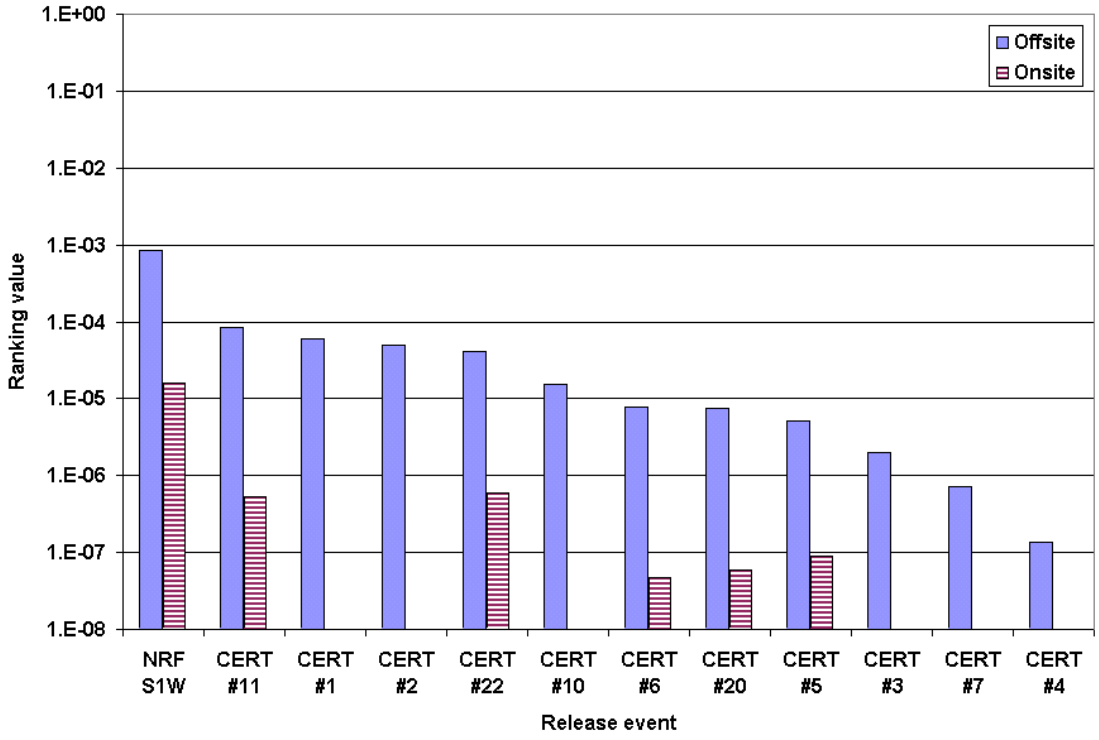


Figure 27. Offsite and onsite ranking values for Group 3 release events.

Table 31. Group 3 Relative Ranking Values

	Ranking value	
Release	Offsite	Onsite
NRF S1W	8E-04	2E-05
CERT #11	8E-05	5E-07
CERT #1	6E-05	na
CERT #2	5E-05	na
CERT #22	4E-05	6E-07
CERT #10	2E-05	na
CERT #6	8E-06	5E-08
CERT #20	7E-06	6E-08
CERT #5	5E-06	9E-08
CERT #3	2E-06	na
CERT #7	7E-07	na
CERT #4	1E-07	na

Group 4 Evaluation

As discussed previously, Group 4 consists of release events occurring over the course of several days or more, for which known quantities of radionuclides were released, and that DOE (1991a) did not evaluate explicitly as episodic releases and therefore did not report dispersion factors. We estimated offsite and onsite (where appropriate) dispersion factors for each release as discussed previously. Tables 32a and 32b show release-specific variables for each release event evaluated as part of this group. At the end of this section, relative ranking values are given for each Group 4 release event.

Experimental Release Tests (Group 4, #1-11)

There were a number of intentional release tests at the INEEL designed to study various parameters related to the movement of radionuclides in the environment. These release tests are briefly described in the sections below.

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. Hawley (1964) reports that the CERT No. 1 release occurred near ground level over a 30-minute period. Additional information regarding this test series is provided by Hawley (1964), Bunch (1966 and 1968), and Zimbrick and Voillequé (1969).

Table 33 lists the releases included in this test series, the location and date of each release, the radionuclides and quantities released, and associated references. A number of CERT releases were evaluated specifically as episodic releases by DOE (1991a) and are assessed as part of the Group 3 releases. Additional CERT releases were selected for evaluation as part of the Group 4 release events so that the releases consisting of the largest amount of each radionuclide were considered.

Table 32a. Release-specific Variables for Group 4 Release Events

Release #	Release event	Date of release	Season[a]	Ingestion[b]	Assumed mixing height (m)	Assumed wind speed (m s^{-1})	Assumed stability class
1	CERT #12	7/26/66	summer		1580	3.7[c]	F
2	CERT #27	11/5/69	autumn	no	940	2.0	F
3	CERT Ce-Cs Release	7/8/76	summer		1580	4.0[c]	F
4	CERT K-42 Release	9/22/70	summer		1580	2.0	F
5	CERT Kr-85 Lab Release	1/20/70	winter	no	565	2.0	F
6	CERT S-35 Release	9/18/68	summer		1580	2.0	F
7	EXCES, Na-5	4/24/70	spring	no	1405	2.0	F
8	EXCES, Xe-4	Aug-69	summer		1580	2.0	F
9	LDDT-1	3/3/71	winter	no	565	5.0[c]	F
10	LDDT-3	9/22/72	summer		1580	5.0[c]	F
11	RDT-4	10/1/69	autumn		940	2.0	F
12	RaLa daily release	5/28/58	spring		1405	2.0	F
13	Ru release	10/30/88	autumn		940	2.0	F
14	Ru-Cs release (1972)	1/9/72	winter	no	565	2.0	F
15	TSF Evaporator release	9/1/71	summer		1580	2.0	F
16	EBR-II Na release and fire	2/1/68	winter	no	565	2.0	F
17	ETR fuel melt incident (sight glass incident)	12/12/61	autumn	no	940	2.0	F
18	OMRE solvent burning experiment	11/16/60	autumn	no	940	11.0[c]	C[d]
19	Collection Tank release	12/1/58	autumn	no	940	2.0	F
20	Solvent burner release	9/1/58	summer		1580	2.0	F
21	MTR stack release-1	12/17/58	autumn	no	940	2.0	F

Release #	Release event	Date of release	Season[a]	Ingestion[b]	Assumed mixing height (m)	Assumed wind speed (m s^{-1})	Assumed stability class
22	MTR stack release-2	2/22/63	winter	no	565	2.0	F
23	WCF ruthenium release	10/16/64	autumn		940	2.0	F
24	Iodine release	10/31/66	autumn		940	2.0	F
25	Xe-133 release from ECF	11/7/66	autumn	no	940	2.0	F
26	ICPP release	6/26/74	summer		1580	2.0	F
27	Unplanned releases - EBRII	1/8/77	winter	no	565	2.0	F
28	Unplanned releases - ICPP	12/14/77	autumn	no	940	2.0	F
29	Release due to APS failure	11/14/77	autumn	no	940	2.0	F
30	Kr-85 release	2/22/81	winter	no	565	2.0	F
31	Ru-Cs release (1984)	6/25/84	summer		1580	2.0	F
32	Blower failure	8/1/58	summer		1580	2.0	F
33	Iodine-129 Technology Studies	8/1/64	summer		1580	2.0	F

[a] Spring: March 21 - June 20, Summer: June 21 - September 22, Autumn: September 23 - December 21, Winter: December 22 - March 20

[b] If release occurs between November and April, assumed no ingestion pathway contribution

[c] Wind speed based on available documentation specific to release

[d] Stability based on available documentation specific to release

Table 32b. Release-specific Variables for Group 4 Release Events

				Offsite			Onsite		
Release #	Release event	Release Location	Release height (m)	Location[a]	Downwind distance (km)	X/Q (s m^{-3})	Location	Downwind distance (km)	X/Q (s m^{-3})
1	CERT #12	EFS	0	Building	44	1.13E-06	HW 33	35	1.46E-06
2	CERT #27	EFS	0	Building	44	2.08E-06	HW 33	35	2.71E-06
3	CERT Ce-Cs Release	NE of NRF	0	Building	38	1.23E-06	HW 33	29	1.68E-06
4	CERT K-42 Release	EFS	0	Building	44	2.08E-06	HW 33	35	2.71E-06
5	CERT Kr-85 Lab Release	CFA	0	Mud Lake	49	1.84E-06	na		
6	CERT S-35 Release	EFS	0	Building	44	2.08E-06	HW 33	35	2.71E-06
7	EXCES, Na-5	GRID III	0	Building	45	2.03E-06	HW 33	36	2.62E-06
8	EXCES, Xe-4	GRID III	0	Building	45	2.03E-06	HW 33	36	2.62E-06
9	LDDT-1	ICPP	0	Building	47	7.73E-07	HW 33	38	9.86E-07
10	LDDT-3	ICPP	0	Building	47	7.73E-07	HW 33	38	9.86E-07
11	RDT-4	GRID III	0	Building	45	2.03E-06	HW 33	36	2.62E-06
12	RaLa daily release	ICPP	76	Atomic City	20	2.38E-06	HW 20	6	3.11E-06
13	Ru release	ICPP	76	Atomic City	20	2.38E-06	HW 20	6	3.11E-06
14	Ru-Cs release (1972)	ICPP	76	Atomic City	20	2.38E-06	HW 20	6	3.11E-06
15	TSF Evaporator release	TSF/TAN	51	Building	14	5.18E-06	HW 33	5	1.05E-05
16	EBR-II Na release and fire	EBR-II	61	Atomic City	20	3.16E-06	HW 20	5	6.52E-06
17	ETR fuel melt incident (sight glass incident)	ETR	76	Frenchman's Cabin	20	2.38E-06	HW 20	5	2.71E-06
18	OMRE solvent burning experiment	OMRE	0	Cerro Grande	10	2.31E-06	HW 20	5	5.81E-06
19	Collection Tank release	ICPP	0	Atomic City	20	5.27E-06	HW 20	6	2.49E-05
20	Solvent burner release	ICPP	76	Atomic City	20	2.38E-06	HW 20	6	3.11E-06
21	MTR stack release-1	MTR	76	Frenchman's Cabin	20	2.38E-06	HW 20	5	2.71E-06

				Offsite			Onsite		
Release #	Release event	Release Location	Release height (m)	Location[a]	Downwind distance (km)	X/Q (s m^{-3})	Location	Downwind distance (km)	X/Q (s m^{-3})
22	MTR stack release-2	MTR	76	Frenchman's Cabin	20	2.38E-06	HW 20	5	2.71E-06
23	WCF ruthenium release	WCF (ICPP)	76	Atomic City	20	2.38E-06	HW 20	6	3.11E-06
24	Iodine release	ICPP	76	Atomic City	20	2.38E-06	HW 20	6	3.11E-06
25	Xe-133 release from ECF	ECF	25	Howe	16	6.26E-06	na		
26	ICPP release	ICPP	76	Atomic City	20	2.38E-06	HW 20	6	3.11E-06
27	Unplanned releases - EBRII	EBR II (ANL)	61	Atomic City	20	3.16E-06	HW 20	5	6.52E-06
28	Unplanned releases - ICPP	ICPP	76	Atomic City	20	2.38E-06	HW 20	6	3.11E-06
29	Release due to APS failure	ICPP	76	Atomic City	20	2.38E-06	HW 20	6	3.11E-06
30	Kr-85 release	ICPP	76	Atomic City	20	2.38E-06	HW 20	6	3.11E-06
31	Ru-Cs release (1984)	CSSF bin/ICPP	0	Atomic City	20	5.27E-06	HW 20	6	2.49E-05
32	Blower failure	ICPP	76	Atomic City	20	2.38E-06	HW 20	6	3.11E-06
33	Iodine-129 Technology Studies	GRID III	0	Building	45	2.03E-06	HW 33	36	2.62E-06

[a] See Figure 21

na – onsite exposure scenario was not applicable because the trajectory of the release did not result in the plume passing over an onsite publicly accessible highway

Table 33. Summary Information for CERT Releases

Release	Location	Date	Curies	Radionuclide	References
CERT #1[a]	N of Atomic City	5/27/63	~1	I-131	Hawley (1964), Bunch (1968)
CERT #2[a]	EFS	9/2/64	~1	I-131	Bunch (1968)
CERT #3[a]	EFS	12/11/64	1	I-131	Bunch (1968)
CERT #4[a]	ARA	5/27/65	0.01	I-131	Bunch (1968)
CERT #5[a]	ARA	6/10/65	0.1	I-131	Bunch (1968)
CERT #6[a]	ICPP	9/14/65	6	I-131[c]	Bunch (1966), Bunch (1968)
CERT #7[a]	EFS	11/22/65	1.1	I-131	Bunch (1966), Bunch (1968)
CERT #8	NE ICPP	5/31/66	0.05	I-131	Bunch (1968)
CERT #9	NE ICPP	6/7/66	0.05	I-131	Bunch (1968)
CERT #10[a]	GRID III	6/14/66	5	I-131	Bunch (1968)
CERT #11[a]	EFS	7/21/66	8	I-131[c]	Bunch (1968)
CERT #12[b]	EFS	7/26/66	1	I-131	Bunch (1968)
CERT #13	EFS	8/3/66	0.1	I-131	Bunch (1968)
CERT #14	EFS	8/5/66	0.1	I-131	Bunch (1968)
CERT #15	EFS	8/5/66	0.1	I-131	Bunch (1968)
CERT #16	EFS	8/24/66	0.1	I-131	Bunch (1968)
CERT #17	EFS	8/24/66	0.1	I-131	Bunch (1968)
CERT #18	EFS	9/8/66	0.1	I-131	Bunch (1968)
CERT #19	EFS	11/7/66	0.5	I-131	Bunch (1968)
CERT #20[a]	EFS	7/6/67	0.9	I-131	Zimbrick and Voilleque (1969)
CERT #22[a]	EFS	9/22/67	1	I-131	Zimbrick and Voilleque (1969)
CERT #23	EFS	6/17/68	0.5	I-131	AEC (1969)
CERT #24	EFS	8/15/68	0.5	I-131	AEC (1969)
CERT #27[b]	EFS	11/5/69	0.5	Cr-51	AEC (1970)
CERT Ce and Cs Release[b]	NE of NRF	7/8/76	0.6	Ce-141, Cs-134	DOE (1978)
CERT Ce and Cs Release	NE of NRF	6/6/75	0.5	Ce-141, Cs-134	DOE (1978)
CERT Cesium Release	NE of NRF	Dec-77	0.1	Cs-134	DOE (1978)
CERT K-42 Release	EFS	10/21/70	0.1	K-42	AEC (1971)
CERT K-42 Release[b]	EFS	9/22/70	0.3	K-42	AEC (1971)
CERT Kr-85 Lab Release[b]	CFA	1/20/70	2	Kr-85	AEC (1971)
CERT S-35 Release[b]	EFS	9/18/68	0.3	S-35	AEC (1969)

[a] Evaluated with Group 3 releases
[b] Evaluated with Group 4 releases
[c] Methyl iodide

Relative Diffusion Tests

The Relative Diffusion Test (RDT) 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).

Table 34 lists the releases included in this test series, the location and date of each release, the radionuclides and quantities released, and associated references. The RDT with the largest associated release was selected for evaluation as part of the Group 4 release events.

Table 34. Summary Information for RDT Releases

Test	Location	Date	Curies	Radionuclide	References
RDT-1	GRID III	11/30/67	1.2	I-131	DOE (1991b)
RDT-2	GRID III	5/7/68	1	I-131	AEC (1969), DOE (1991b)
RDT-3	GRID III	5/7/68	1.8	I-131[b]	AEC (1969), DOE (1991b)
RDT-4[a]	GRID III	10/1/69	6	I-131[c]	DOE (1991b)

[a] Evaluated with Group 4 releases
[b] 1 Ci methyl, 0.84 Ci elemental
[c] 5 Ci methyl, 1 Ci elemental

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.

Table 35 lists the releases included in this test series, the location and date of each release, the radionuclides and quantities released, and associated references. The tests with the largest associated release of each radionuclide were selected for evaluation as part of the Group 4 release events.

Table 35. Summary Information for EXCES Releases

Release	Location	Date	Curies	Radionuclide	References
EXCES, Na-1	GRID III	3/25/70	51	Na-24	AEC (1971)
EXCES, Na-2	GRID III	4/1/70	75	Na-24	AEC (1971)
EXCES, Na-3	GRID III	4/7/70	6.6	Na-24	AEC (1971)
EXCES, Na-4	GRID III	4/9/70	103	Na-24	AEC (1971)
EXCES, Na-5[a]	GRID III	4/24/70	120	Na-24	AEC (1971)
EXCES, Xe-2	GRID III	5/3/68	32	Xe-133	AEC (1969)
EXCES, Xe-3	GRID III	Aug-69	300	Xe-133	AEC (1970)
EXCES, Xe-4[a]	GRID III	Aug-69	300	Xe-133	AEC (1970)

[a] Evaluated with Group 4 releases

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.

Table 36 lists the releases included in this test series, the location and date of each release, the radionuclides and quantities released, and associated references. The test with the largest associated release was selected for evaluation as part of the Group 4 release events.

Table 36. Summary Information for LDDT Releases

Release	Location	Start date	Curies	radionuclides	References
LDDT-1	ICPP	3/3/71	4.4	I-131 (methyl)	Voilleque (1971), DOE (1991a)
LDDT-2	ICPP	8/31/71	4.2	I-131 (methyl)	Voilleque (1971), DOE (1991a)
LDDT-3[a]	ICPP	9/22/72	1000	Kr-85	Dickson and Voilleque (1972), DOE (1991a)
			3.7	I-131 (methyl)	

[a] Evaluated with Group 4 releases

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. 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.

RaLa Daily Release (Group 4, #12)

We evaluated the highest daily release of ^{131}I, which occurred on May 28, 1958, and amounted to 49.5 Ci. Significant quantities of ^{131}I were released on a number of other days; those days where reported releases exceeded 30 Ci are shown in Table 37.

Table 37. Daily RaLa Releases Exceeding 30 Ci

Day	Curies	Radionuclide
5/28/58	49.5	^{131}I
3/1/58	48.2	^{131}I
10/7/57	40.6	^{131}I
5/13/58	39.3	^{131}I
6/10/58	34.2	^{131}I
5/29/58	33.4	^{131}I
6/11/58	32.5	^{131}I
2/6/59	31.5	^{131}I
11/8/57	31.4	^{131}I

1988 Ruthenium Release (Group 4, #13)

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).

1972 Particulate Release (Group 4, #14)

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.

TSF Evaporator Release (Group 4, #15)

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 assumed the release consisted entirely of ^{137}Cs.

EBR-II Sodium Release (Group 4, #16)

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.

ETR Sight Glass Incident (Group 4, #17)

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). We assumed the particulate activity was composed entirely of ^{138}Cs and the fission gas activity consisted entirely of ^{88}Kr. 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. 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.

Organic Moderated Reactor Experiment Solvent Burning Experiment (Group 4, #18)

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 assumed that the entire 400 gallons had the radioactivity concentration

reported for xylene and that ^{60}Co was the only radionuclide present. If everything present was volatilized, a total of 0.026 Ci could have been released to the atmosphere.

Collection Tank Release (Group 4, #19)

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 assumed the release was comprised entirely of ^{131}I.

Solvent Burner Release (Group 4, #20)

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 assumed the release was comprised entirely of ^{137}Cs.

MTR Stack Release (Group 4, #21)

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 assumed the fission gas activity consisted entirely of ^{88}Kr.

MTR Stack Release (Group 4, #22)

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 assumed the fission gas activity consisted entirely of ^{88}Kr.

Daily WCF Ruthenium Releases (Group 4, #23)

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 releases occurring on a single day, we selected the highest daily release (54.6 Ci on October 16, 1964). 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).

ICPP Iodine Release (Group 4, #24)

A total of 1.48 Ci radioiodine was released from the ICPP on October 31, 1966 (Horan 1966). We assumed the release consisted entirely of ^{131}I.

ECF Iodine Release (Group 4, #25)

A radioactive release occurred at the Expended Core Facility (ECF) at the NRF on November 7, 1966. The release was reported to consist of 2.8 Ci ^{133}Xe (Abrams 1966).

ICPP Release (Group 4, #26)

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 $^{144}Ce/Pr$.

1977 Unplanned Releases (Group 4, #27 and #28)

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 assumed the entire cask release escaped the building and was emitted to the atmosphere.

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.

Release due to APS Failure (Group 4, #29)

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, so we assumed the release consisted entirely of ^{137}Cs.

ICPP Release (Group 4, #30)

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).

Calcine Bin Vent Filter Failure Release (Group 4, #31)

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.

Blower Failure at ICPP (Group 4, #32)

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 assumed the release was comprised entirely of ^{137}Cs.

Iodine-129 Technology Studies Release (Group 4, #33)

In August 1964, a total of five tests were conducted to examine atmospheric mixing and dilution of gases and particles containing small amounts of ^{129}I (INEEL 1995). Less than one millicurie of ^{129}I was released as part of the experiment. We assumed a total of one millicurie was released on a single day for this evaluation because the time period over which the tests occurred was not specified. We assumed the release took place at GRID III, although the location of the tests was not specified.

Group 4 Ranking Results

Figure 28 provides a graphical representation of the offsite and onsite ranking values for each Group 4 release, sorted from highest to lowest according to the offsite ranking values (Table 38). As with the other groups of releases, there are clearly differences in the overall offsite and onsite ranking orders. These differences are again related to the different onsite and offsite distances estimated for each release event, the different radionuclides comprising the release, and the fact that onsite exposure scenarios consider the inhalation and immersion pathways only. For Release #19, #8, and #30, the onsite ranking value is higher than the offsite value. This results partly from the higher dispersion factor for the onsite location. This also occurs for Release #19 because the ingestion pathway was not considered for the offsite exposure scenario, and for Releases #8 and #30 because the radionuclides involved with those releases are important for the immersion pathway only. Onsite ranking values were not estimated for Release #5 or #25 because the trajectory of the release did not result in the plume passing over an onsite publicly accessible highway.

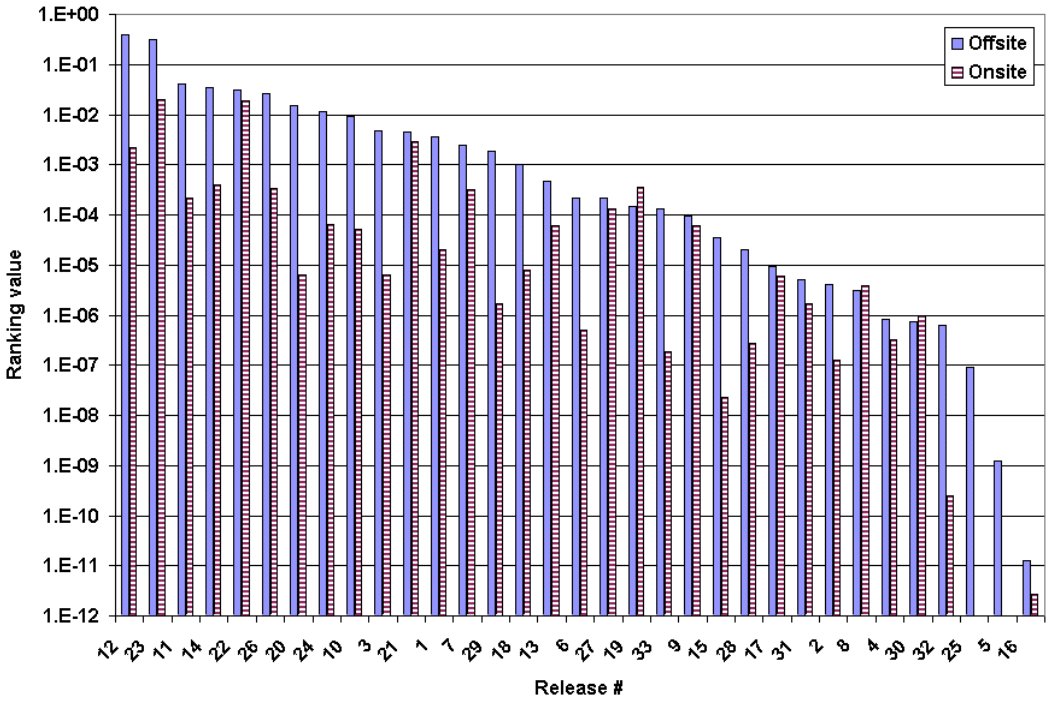


Figure 28. Offsite and onsite ranking values for Group 4 release events.

For the Group 4 releases, Releases #12 (RaLa Daily Release) and #23 (WCF Ru Release) are clearly the highest-ranking release with regard to the offsite exposure scenario. Releases #11 (RDT-4), #14 (1972 Ru-Cs Release), #22 (MTR Stack Release-2), and #26 (1974 ICPP Release) are also relatively high ranking. The importance of Release #11 is likely over-estimated because it consisted primarily 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 ranking values calculated assuming the releases consisted of elemental radioiodine are likely erroneously high. The importance of Release #14 may also be overestimated, based on the high NCRP/RSAC ratio for ^{137}Cs when the ingestion pathway is not considered, as discussed in Appendix B. The importance of Release #22 may also be overestimated by assuming the gaseous fission product release consisted entirely of ^{88}Kr.

Table 38. Group 4 Relative Ranking Values

		Ranking value	
Release #	Release	Offsite	Onsite
12	RaLa daily release	4.E-01	2.E-03
23	WCF ruthenium release	3.E-01	2.E-02
11	RDT-4	4.E-02	2.E-04
14	Ru-Cs release (1972)	3.E-02	4.E-04
22	MTR stack release-2	3.E-02	2.E-02
26	ICPP release	3.E-02	3.E-04
20	Solvent burner release	2.E-02	6.E-06
24	Iodine release	1.E-02	6.E-05
10	LDDT-3	9.E-03	5.E-05
3	CERT Ce-Cs Release	5.E-03	6.E-06
21	MTR stack release-1	4.E-03	3.E-03
1	CERT #12	4.E-03	2.E-05
7	EXCES, Na-5	2.E-03	3.E-04
29	Release due to APS failure	2.E-03	2.E-06
18	OMRE solvent burning experiment	1.E-03	8.E-06
13	Ru release	5.E-04	6.E-05
6	CERT S-35 Release	2.E-04	5.E-07
27	Unplanned releases - EBRII	2.E-04	1.E-04
19	Collection Tank release	1.E-04	4.E-04
33	Iodine-129 Technology Studies	1.E-04	2.E-07
9	LDDT-1	1.E-04	6.E-05
15	TSF Evaporator release	4.E-05	2.E-08
28	Unplanned releases - ICPP	2.E-05	3.E-07
17	ETR fuel melt incident (sight glass incident)	9.E-06	6.E-06
31	Ru-Cs release (1984)	5.E-06	2.E-06
2	CERT #27	4.E-06	1.E-07
8	EXCES, Xe-4	3.E-06	4.E-06
4	CERT K-42 Release	8.E-07	3.E-07
30	Kr-85 release	7.E-07	1.E-06
32	Blower failure	6.E-07	3.E-10
25	Xe-133 release from ECF	9.E-08	
5	CERT Kr-85 Lab Release	1.E-09	
16	EBR-II Na release and fire	1.E-11	3.E-12

Highest onsite ranking values are highlighted

With regard to onsite exposure, Releases #23, #22, and #12 also appear to be most important, along with Release #21 (MTR Stack Release-1). Again, the importance of Release #21 and #22 may be over-estimated by assuming the gaseous fission product release in both cases consisted entirely of ^{88}Kr.

Although the ranking values for the Group 4 releases are not directly comparable to ranking values established for releases in other groups because of differences in the assumptions required to evaluate the releases in each group, some general conclusions can be drawn based on the magnitude of the ranking values. Because the highest ranking values for the Group 4 releases are similar to the highest ranking values for either the Group 1 or Group 2 releases, it is possible that their overall relative importance in terms of health impacts to exposed individuals is also similar. However, a comparison of the dispersion factors estimated for the Group 2 (short-duration) releases (Table 21b) to the dispersion factors estimated for the Group 4 (short-duration) releases (Table 32b) suggests that Group 4 ranking values would likely be less if release-specific meteorological conditions were considered. As a result, the first step for any further evaluation of a Group 4 release event should be to refine the atmospheric dispersion factor for the release, based on meteorological conditions existing at the time of the release.

Group 5 Evaluation

As discussed previously, Group 5 consists of release events occurring during a period of one day or less, for which known quantities of radionuclides were released, and that DOE (1991a) did not evaluate explicitly as episodic releases and therefore did not report dispersion factors. We estimated offsite and onsite (where appropriate) dispersion factors for each release as discussed previously. Tables 39a and 39b show release-specific variables for each release event evaluated as part of this group. At the end of this section, relative ranking values are given for each Group 5 release event.

Table 39a. Release-specific Variables for Group 5 Release Events

Release #	Release event	Dates of release		Season[a]	Ingestion[b]	Assumed mixing height (m)	Assumed wind speed ($m\ s^{-1}$)	Assumed stability class
		Start	Stop					
1	RaLa Run #2	2/24/57	3/30/57	winter	no	565	2.0	D
2	RaLa Run #8	10/21/57	1/5/58	autumn/winter	no	750	2.0	D
3	RaLa Run #1	2/3/57	2/23/57	winter	no	565	2.0	D
4	RaLa Run #15	6/2/58	8/5/58	spring/summer		1490	2.0	D
5	RaLa Run #7	10/7/57	10/20/57	autumn		940	2.0	D
6	LOFT LP-FP-2 Test	7/9/85	9/9/85	summer		1580	2.0	D
7	ICPP Pu release	7/9/59	7/11/59	summer		1580	2.0	D
8	MTR stack release	4/30/60	5/6/60	spring		1405	2.0	D
9	ETR stack release	6/20/60	6/21/60	spring/summer		1490	2.0	D
10	WCF ruthenium releases	10/1/64	10/31/64	autumn		940	2.0	D
11	Kr-85 and Sb-125 releases	9/27/74	10/27/74	autumn		940	2.0	D
12	WCF releases	5/1/72	5/31/72	spring		1405	2.0	D

[a] Spring: March 21 - June 20, Summer: June 21 - September 22, Autumn: September 23 - December 21, Winter: December 22 - March 20

[b] If release occurs between November and April, assumed no ingestion pathway contribution

Table 39b. Release-specific Variables for Group 5 Release Events

				Offsite			Onsite		
Release #	Release event	Release Location	Release height (m)	Location[a]	Downwind distance (km)	X/Q (s m^{-3})	Location	Downwind distance (km)	X/Q (s m^{-3})
1	RaLa Run #2	ICPP	76	Atomic City	20	7.39E-07	HW 20	6	3.48E-06
2	RaLa Run #8	ICPP	76	Atomic City	20	7.39E-07	HW 20	6	3.48E-06
3	RaLa Run #1	ICPP	76	Atomic City	20	7.39E-07	HW 20	6	3.48E-06
4	RaLa Run #15	ICPP	76	Atomic City	20	7.39E-07	HW 20	6	3.48E-06
5	RaLa Run #7	ICPP	76	Atomic City	20	7.39E-07	HW 20	6	3.48E-06
6	LOFT LP-FP-2 Test	LOFT (TAN)	46	Building	14	1.28E-06	HW 33	5	5.37E-06
7	ICPP Pu release	ICPP	6	Atomic City	20	7.94E-07	HW 20	6	4.66E-06
8	MTR stack release	MTR	76	Frenchman's Cabin	20	7.39E-07	HW 20	5	4.25E-06
9	ETR stack release	ETR	76	Frenchman's Cabin	20	7.39E-07	HW 20	5	4.25E-06
10	WCF ruthenium releases	WCF (ICPP)	76	Atomic City	20	7.39E-07	HW 20	6	3.48E-06
12	Kr-85 and Sb-125 releases	ICPP	76	Atomic City	20	7.39E-07	HW 20	6	3.48E-06
13	WCF releases	WCF (ICPP)	76	Atomic City	20	7.39E-07	HW 20	6	3.48E-06

[a] See Figure 21

RaLa Run Releases (Group 5; #1, 2, 3, 4, and 5)

Releases associated with the RaLa program were discussed in the RaLa Iodine Releases section in the Group 4 evaluation section. We also evaluated the highest total releases associated with specific RaLa runs, which occurred over a period of several days or more, as part of Group 5 (Releases #1, 2, 3, 4, and 5). 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. Significant quantities of ^{131}I were released during several other RaLa runs; those runs where reported releases exceeded 200 Ci are shown in Table 40.

Table 40. RaLa Runs with Releases Exceeding 200 Ci

Run	Week of	Curies	Radionuclide
RaLa Run 002	2/24/57	351.3	^{131}I
RaLa Run 008	10/21/57	310.9	^{131}I
RaLa Run 001	2/3/57	252.9	^{131}I
RaLa Run 015	6/2/58	205.8	^{131}I
RaLa Run 007	10/7/57	201.7	^{131}I

LOFT LP-FP-2 Test (Group 5, #6)

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 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.

ICPP Plutonium Release (Group 5, #7)

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 ft 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 ft from the burner building.

MTR Stack Release (Group 5, #8)

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 assumed the fission gas activity consisted entirely of ^{88}Kr.

ETR Stack Release (Group 5, #9)

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 assumed the released activity consisted entirely of ^{138}Cs, the radionuclide with the largest screening factor.

Monthly WCF Ruthenium Releases (Group 5, #10)

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 releases occurring over the course of a month, we selected the highest monthly release (96.8 Ci in October 1964). 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).

ICPP Releases (Group 5, #11)

Releases of ^{85}Kr and ^{125}Sb were 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.

WCF Release (Group 5, #12)

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. To estimate the composition, we divided the release quantities derived for the FECF Release (see Group 2 Evaluation) by 5.4 to derive a total release of approximately 20 Ci.

Group 5 Ranking Results

Figure 29 provides a graphical representation of the offsite and onsite ranking values for each Group 4 release, sorted from highest to lowest according to the offsite ranking values (Table 41). As with the other groups of releases, there are clearly differences in the overall offsite and onsite ranking orders. These differences are again related to the different onsite and offsite distances estimated for each release event, the different radionuclides comprising the release, and the fact that the onsite exposure scenarios consider the inhalation and immersion pathways only. For several releases, the onsite ranking value is higher than the offsite value. As with the Group 4 releases, this results partly from the higher dispersion factor for the onsite location and also is related to the relatively greater importance of the inhalation and immersion pathways for the radionuclides comprising these releases.

For the Group 5 releases, Releases #4 (RaLa Run #15), #5 (RaLa Run #7), and #10 (WCF Ru Releases) are clearly the highest-ranking releases with regard to the offsite exposure scenario. The overall importance of the ingestion pathway is demonstrated by the lower ranking values for Releases #1 (RaLa Run #2), #2 (RaLa Run #8), and #3 (RaLa Run #1), which all involved greater amounts of ^{131}I than either Release #4 or #5, but occurred during times of the year when the ingestion pathway was not important to consider.

With regard to onsite exposure, Releases #10 and #7 (ICPP Pu release) are highest ranking, with several releases of slightly lesser importance (e.g., Releases #6, #1, and #2).

Although the ranking values for the Group 5 releases are not directly comparable to ranking values established for releases in other groups because of differences in the assumptions required to evaluate the releases in each group, some general conclusions can be drawn based on the magnitude of the ranking values. Because the highest ranking values for the Group 5 releases are similar to the highest ranking values for either the Group 1 or Group 2 releases, it is possible that their overall relative importance in terms of health impacts to exposed individuals is also similar. However, a comparison of the dispersion factors estimated for the Group 1 (longer-duration) releases (Table 18b) to the dispersion factors estimated for the Group 5 (longer-duration) releases (Table 39b) suggests that Group 5 ranking values would likely be less if release-specific meteorological conditions were considered. As a result, as with the Group 4 releases, the first step for any further evaluation of a Group 5 release event should be to refine the atmospheric dispersion factor for the release, based on meteorological conditions existing at the time of the release.

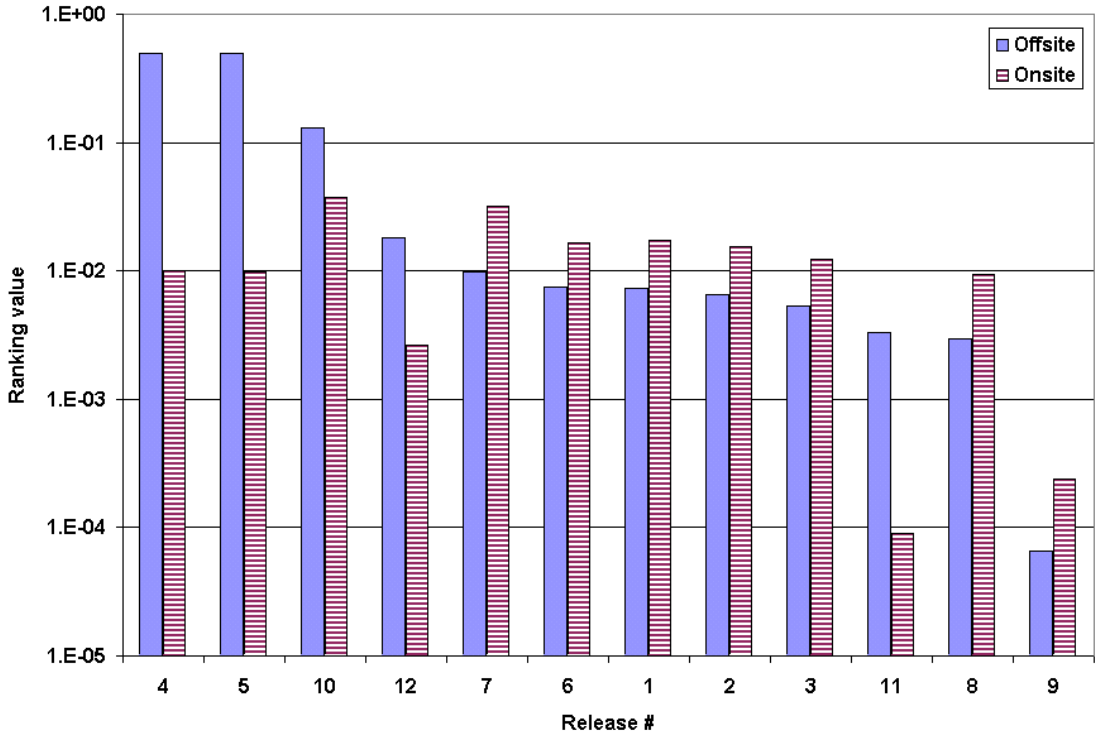


Figure 29. Offsite and onsite ranking values for Group 5 release events.

Table 41. Group 5 Relative Ranking Values

		Ranking value	
Release #	Release	Offsite	Onsite
4	RaLa Run #15	5.E-01	1.E-02
5	RaLa Run #7	5.E-01	1.E-02
10	WCF ruthenium releases	1.E-01	4.E-02
12	WCF releases	2.E-02	3.E-03
7	ICPP Pu release	1.E-02	3.E-02
6	LOFT LP-FP-2 Test	8.E-03	2.E-02
1	RaLa Run #2	7.E-03	2.E-02
2	RaLa Run #8	6.E-03	2.E-02
3	RaLa Run #1	5.E-03	1.E-02
11	Kr-85 and Sb-125 releases	3.E-03	9.E-05
8	MTR stack release	3.E-03	9.E-03
9	ETR stack release	7.E-05	2.E-04

Highest onsite ranking values are highlighted

Releases Not Evaluated

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.

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.

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). This release was not evaluated as a separate episodic release because other similar and larger releases have already been evaluated and specifics regarding the composition of the release were not provided. 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.

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 we evaluated in detail.

Limitations of this Evaluation

The scope of work and resources available for this project did not enable a detailed reconstruction of all episodic releases. As discussed previously, this was not the intent of the project nor would it be an efficient expenditure of resources at this stage of an analysis of releases from the INEEL. As a result, some simplifying assumptions were necessary. However, the required task of prioritizing episodic release events necessitated a more realistic evaluation of all releases than simple screening typically employs. To the extent possible, this more realistic approach reduces the possibility of introducing inconsistent biases that could significantly impact the ranking results for episodic releases. We believe the approach taken for this work has accomplished this goal; however, some aspects of the analyses are still relatively simplistic and should not be considered to be commensurate with a detailed dose reconstruction effort. Accounting for every process and factor ultimately contributing to the actual dose or risk to an exposed individual is a complex process that would be part of a more detailed analysis. Nevertheless, the data compiled and analyses presented in this report provide a solid benchmark for guiding and focusing any additional work that may be completed for the INEEL.

Should additional more detailed analyses of any of these release events be undertaken, further refinement of some assumptions and parameters will be required.

- In particular, the precise timing of longer duration events, such as the releases associated with the IET program, may require more detailed investigation, and meteorological refinements corresponding to the timing of the release based on additional review of available historical data may be necessary.
- The adequacy of measurement techniques used to determine gross activity release estimates for the IET series will likely require additional investigation, and the discussion included in DOE (1991a) should serve as a starting point.
- The gross activity release estimates based on maximum reported release rates for a number of the IET release events are likely very conservative in most cases and may require additional investigation to estimate a more realistic gross activity release.
- The release fractions assumed for this analysis should be carefully reviewed to determine whether it is possible to derive values that may be more realistic based on detailed review of the characteristics of the release. Determinations made as part of the FPFRT and FEBT releases may assist with refining the release fractions assumed for the IET releases. In addition, the assumptions made by DOE (1991a) with regard to release fraction estimation should be reviewed to determine if the methodology developed as part of that work could assist with any more detailed analyses that are deemed necessary.
- The potential impact of effluent temperature on plume rise may also require further investigation because only the physical stack heights for each release were considered as part of this current analysis.
- The parameters driving deposition of radionuclides on the ground surface and subsequent transfer through the food chain will need to be carefully examined as part of a more detailed analysis and appropriate values determined for each release event.
- Existing historical environmental monitoring data should be utilized wherever possible to validate the accuracy of model-predicted concentrations
- It was not possible to derive release specific atmospheric dispersion factors for most of the Group 4 and 5 release events. As a result, the first step for any further evaluation of these release events should be to refine the atmospheric dispersion factor for the release, based on meteorological conditions existing at the time of the release.

Conclusions and Summary of Relative Ranking Results for Episodic Release Events

We calculated specific ranking values for a total of 99 individual release events, which we segregated into 5 different groups. Our searches and reviews of the Task Order 6 database enabled us to identify a number of release events that had not been previously identified or evaluated by DOE (1991a). We attempted to evaluate all of the episodic releases as consistently and realistically as possible, considering the information available to characterize each release.

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.

Based on the close agreement between the ranking orders developed using both methodologies for the Group 1 and Group 2 release events and the more detailed analysis presented in Appendix B, we conclude that the NCRP screening factors are appropriate for understanding the relative importance of episodic or short-term releases. Furthermore, the general agreement between the NCRP and RSAC methodologies suggests that the NCRP screening factors may also be suitable in many cases for estimating potential doses for short-duration releases.

It is important to note that differences between the NCRP and RSAC methodologies when all pathways are considered (i.e., "Total") are relatively small, as evidenced by the data presented in Figure B5 and the ratios presented in the "Total" columns of both Tables B3 and B4. This consistent agreement between the two methodologies suggests that significant discrepancies between the two methodologies are generally confined to exposure pathways that are not major contributors to the sum of all pathways. The consistent agreement between ranking values estimated by the two methodologies for the Group 1 and Group 2 release events, most of which were comprised of a number of different radionuclides, suggests that significant discrepancies are also primarily confined to radionuclides that are not major contributors to the overall importance of a release that includes a mixture of fission products.

To summarize the ranking results for the releases evaluated as part of this work, we compiled the releases with the highest relative ranking values in each group based on the NCRP methodology in Table 42. The most realistic evaluations were possible for the Group 1, 2, and 3 release events because release-specific dispersion factors were available for each release. The Group 4 and 5 ranking values may be biased high to some degree, based on the consistently higher dispersion factors for these releases than the release-specific dispersion factors for the Group 1, 2, and 3 releases. As a result, any further evaluation of Group 4 or 5 release events should first focus on refining the dispersion factors for the releases that are determined to warrant additional investigation. The onsite ranking values for the Group 1 and Group 5 release events (and the NRF S1W Engineering Test) are also likely biased high because the assumption of an individual being present at the onsite location throughout the duration of the release becomes less realistic for these longer-term releases.

Combined, the release events highlighted in Table 42 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 for both onsite and offsite scnarios, so a detailed analysis of one or more of these events would allow for an assessment of its potential importance at both onsite and offsite locations. 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. The ranking orders presented in this report can serve as a defensible basis for guiding and focusing future work that may be done with regard to INEEL historical episodic releases.

Table 42. Highest Relative Ranking Values for Episodic Releases

Release group	Offsite			Onsite		
	Release #[a]	Release	RRV[b]	Release #[a]	Release	RRV
Group 1			1.E-01			1.E-01
			2.E-02		IET-04	6.E-04
		IET-14	2.E-03		IET-21	1.E-04
		IET-20	1.E-03		IET-08	1.E-04
		IET-21	1.E-03		IET-19	1.E-04
		IET-23	6.E-04		IET-26	1.E-04
Group 2			1.E-01			5.E-02
			5.E-02			2.E-02
			3.E-02			8.E-03
			3.E-02			5.E-03
			2.E-02		FECF	2.E-03
			2.E-02		BORAX-I	1.E-04
Group 3			8E-04			2.E-05
		CERT #11	8E-05		CERT #22	6.E-07
		CERT #1	6E-05		CERT #11	5.E-07
		CERT #2	5E-05		CERT #5	9.E-08
		CERT #22	4E-05		CERT #20	6.E-08
		CERT #10	2E-05		CERT #6	5.E-08
Group 4	12		4.E-01	23		2.E-02
	23		3.E-01	22		2.E-02
	11		4.E-02	21		3.E-03
	14	Ru-Cs release (1972)	3.E-02	12	RaLa daily release	2.E-03
	22	MTR stack release-2	3.E-02	14	Ru-Cs release (1972)	4.E-04
	26	ICPP release	3.E-02	19	Collection Tank release	4.E-04
Group 5	4		5.E-01	10		4.E-02
	5		5.E-01	7		3.E-02
	10		1.E-01	1		2.E-02
	12	WCF releases	2.E-02	6		2.E-02
	7	ICPP Pu release	1.E-02	2		2.E-02
	6	LOFT LP-FP-2 Test	8.E-03	3		1.E-02

[a] Group 4 and 5 releases only

[b] Relative ranking value

For each group, releases within an order of magnitude of the highest-ranking release in the group are highlighted (based on the ranking values rounded to a single significant figure)

DUCK HUNTER SCENARIO FOR THE INEEL

In addition to routine releases and episodic release events, we also evaluated hunting and eating waterfowl from the INEEL area as a special exposure scenario in response to concerns that some members of the public may have been exposed to releases from the INEEL by exposure to waterfowl that may have resided on ponds at the INEEL. 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 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). 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.

To address these concerns from some members of the community, we developed a special scenario with the help of the INEEL HES at the September 1999 quarterly meeting. We evaluated three special exposure pathways for the scenario, a pregnant female duck hunter: ingestion of meat, 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. We calculated ingestion doses from ^{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.

We established the exposure parameters and duck hunter behaviors through interaction with the INEEL HES members. The scenario involves a female hunter in her first trimester of pregnancy, who shoots 180 ducks over the course of the duck-hunting season of three months from October through December. The bag limit in Idaho at this time is 6 per day. It was agreed that we are calculating dose to the person, not to the duck. Although the female hunter most likely would hunt along the Snake River or in a wildlife management area away from the INEEL, we assumed that twelve of the 180 ducks she shoots have come from the TRA ponds with no loss of radioactivity from the levels measured in the 1980s. 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 for this special scenario are based on measured radionuclide concentrations in tissues from waterfowl from the TRA ponds (Halford et al. 1981; Markham et al. 1988).

We assumed that 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 her vehicle and put in the trunk or in the bed of a truck. Based on information from hunters in the INEEL HES group, we learned that 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. We further

assumed that the feathers are not washed or sorted but used directly to make a pillow to maximixe her possible exposure and to ensure conservatism. Next, the hunter 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 assume she cooks and eats the ducks within a short period of time.

Most people consume only the breast muscle of ducks but we assumed our hunter ate muscle tissue and liver from all 12 contaminated ducks. Basedon these assumptions, we calculated an ingestion dose 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, ^{137}Cs, ^{134}Cs, ^{75}Se, had the highest concentrations in muscle tissue. We also 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 43 lists the radionuclide concentrations measured previously in waterfowl taken from the TRA ponds. While our primary focus was 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 in the 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 43. Concentrations (pCi g^{-1}) of Radionuclides in Waterfowl (Fresh Weight) from the 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}C decays by beta emission to the metastable ^{137m}Ba ($T_{1/2}$ = 2.6 min), 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, all eaten within a short time.
2. Inhalation of ^{239,240}Pu from radioactivity resuspended from the feathers to air in plucking feathers and from using the pillow.

For ^{134}Cs, ^{75}Se, and ^{131}I, we evaluated the ingestion pathway for ingesting muscle and liver from ducks with the average and maximum concentrations of radionulcides. Table 44 summarizes the annual doses calculated for ^{137}Cs and ^{239,240}Pu. The sections following the table describe the assumptions, methods, and calculations of the doses from each pathway.

Calculation Details

The measured ^{137}Cs and ^{239,240}Pu concentrations in various tissues of ducks collected from the TRA ponds at the INEEL (Table 43) were 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 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 from 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*.

Table 44. 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.

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 ^{134}Cs, ^{75}Se, and ^{131}I are taken from the same source. The results of these calculations are shown in Table 45 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.

Table 45. 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 43).

[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).

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 was 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 ft diameter by × 0.5 ft 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.

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 is 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:

$$f_r = C_a / C_s \quad (13)$$

where,

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

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 assumed 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, we calculated the concentration on the surface of the car seat (0.5 m diameter area) for ^{137}Cs as:

C_s for ^{137}Cs = (6.4 pCi g^{-1} × 72 g feathers)/0.2 m^2
C_s for ^{137}Cs = 2304 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 for ^{137}Cs = (6.4 pCi g^{-1} × 72 g feathers)/0.8 m^2
C_s for ^{137}Cs = 580 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 × 64 pCi g^{-1} × 72 g/pillow)/0.029 m^3 = 1.3×10^5 pCi m^{-3} (1.3×10^{-7} 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 × 260 pCi g^{-1} × 72 g/pillow)/0.029 m^3= 5.2×10^5 pCi m^{-3} (5.2×10^{-7} 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 46 and 47 summarize these values and the resulting doses from inhalation of air with these concentrations of ^{137}Cs and $^{239,240}Pu$.

Table 46. 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 (h per y)	Breathing volume (m^3 y^{-1})	Inhalation dose coefficient (mrem pCi^{-1})	Dose from inhalation (mrem)
Car Seat						
Average ^{137}Cs concentration	0.023	0.9	730	660	1.4×10^{-4}	0.002
Maximum ^{137}Cs concentration	0.092	0.9	730	660	1.4×10^{-4}	0.0085
Easy Chair						
Average ^{137}Cs concentration	0.0058	0.9	1095	985	1.4×10^{-4}	0.0008
Maximum ^{137}Cs concentration	0.023	0.9	1095	985	1.4×10^{-4}	0.0032
Pillow						
Average ^{137}Cs concentration	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).

Table 47. 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 (h 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	5.0×10^{-5}	0.9	730	660	5.9×10^{-2}	0.002
Maximum ^{239,240}Pu concentration	2.0×10^{-4}	0.9	730	660	5.9×10^{-2}	0.008
Easy Chair						
Average ^{239,240}Pu concentration	1.3×10^{-5}	0.9	1095	985	5.9×10^{-2}	0.0008
Maximum ^{239,240}Pu concentration	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).

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 ($T_{1/2}$ = 2.6 min), which in turn decays by emitting a

0.66 MeV gamma. We estimated the radiation exposure from a volume containing uniformly distributed gamma emitting isotope from the effective surface activity after allowing for self absorption within the volume (Cember 1988) using the following equation:

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

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

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

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 a linear absorption coefficient for 0.66 MeV gamma of 0.0035 m^{-1}
x = depth of top layer
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):

$$H = \pi \times \Gamma \times (C_v/\mu)\ (1\text{-}e^{-\mu t})\ \ln[(R^2 + h^2)/h^2] \quad (17)$$

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-h) (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 = 2×10^{-9} Sv h^{-1}, or 2×10^{-7} rem h^{-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). Screening doses from external irradiation from the contaminated car seat, easy chair, and pillow, and from the inhalation of resuspended materials from the feathers contributed less than 1% of the total 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 this special waterfowl pathway may not rank relatively high compared to some of the chronic or episodic releases, it may be an important exposure pathway for some individuals who are hunters in the INEEL region.

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

We ranked the potential impact on human health from routine operational releases and from short-term episodic releases that occurred in the past at the INEEL. For the ranking process, we used the screening methods developed by the National Council on Radiation Protection and Measurements (NCRP). To confirm that the NCRP method was suitable for these ranking purposes, we also made dose calculations using the RSAC code to verify that the NCRP screening method was identifying the radionuclides, events, and years that were most important in terms of public health. Duplicate calculations were carried out for several years of routine releases, and for various episodic events, and the ranking results compared. In all cases, the RSAC code confirmed the results obtained using the NCRP methods.

For routine releases from the INEEL, we evaluated all pathways of exposure in the NCRP methodology and all individual years for 56 radionuclides released between 1952 and 1992 at an offsite location at Atomic City where all pathways of exposure were considered (the point of maximum offsite concentration where someone may have lived), and at an onsite location near Highway 20, where the inhalation and plume immersion pathways would be most important for

exposure. The highest ranked radionuclides 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 focused on the high release years during the late 1950s. At the onsite location, where the inhalation and plume immersion pathways were most important, the highest ranked radionuclides included ^{41}Ar, some krypton and xenon isotopes, ^{144}Ce, ^{90}Sr, ^{106}Ru, and ^{131}I, for most of the early years.

For the episodic events and accidents, we calculated specific ranking values for a total of 99 individual release events, which we segregated into 5 different groups. We divided the episodic release events into 5 groups based on the timing and duration of the event, the source term, and the atmospheric dispersion information. Because the assumptions required and information available to characterize each release within a given group was similar, the potential for introducing inconsistent biases into the calculations was minimized. Some of the episodic release events that had high relative ranking at both onsite and offsite locations were the

- Initial Engine Test-10 (December 1957 through March 1958)
- Initial Engine Test-4 (April to June 1956)
- Fuel Element Burn Test B (1957)
- SL-1 reactor accident (January 1961)
- Ruthenium releases from the Waste Calcining Facility at the ICPP (1964)
- Fuel Element Cutting Facility filter break at the ICPP (October 30, 1956)
- Criticality accident at the ICPP (October 1959)
- RaLa releases on 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.

Historically, 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 was 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, consisted of gaseous and particulate radionuclides generated during routine reactor operation but again in much smaller quantities than at TRA and ICPP.

Reprocessing operations began at the ICPP in 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 ranking values. 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 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 impact 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.

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. Ingesting the duck meat was the most important exposure pathway and ^{137}Cs was the largest contributor to the ingestion dose. 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 evidence of movement offsite in groundwater as a potential exposure pathway. Except for tritium, the groundwater pathway was not considered to be a complete offsite exposure pathway for any radionuclides for this historical screening analysis. This analysis only assessed potential 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 ranking evaluation has identified some potential areas of consideration if additional resources and time were to be focused at the INEEL. 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.

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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	
	A1W	17.7 m discharge height; 16 to 22 $m^3 s^{-1}$ discharge capacity; continuously monitored
	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 $m^3 s^{-1}$ 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)	ICPP	94.5 meters deep; flow intermittent
Liquid (Seepage ponds; leaching pits; cribs)	Argonne National Lab-West	Batch monitored subsurface crib.
	Experimental Breeder Reactor-II	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 $76 \times 152 \times 5.5$ 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

APPENDIX B

COMPARISON OF NCRP AND RSAC METHODOLOGIES

Comparison of NCRP and RSAC Methodologies

Based on the results of the Group 1 and Group 2 analyses, the NCRP and RSAC methodologies produce comparable relative ranking results. This is particularly true for releases consisting of a large number of different radionuclides; however, some larger discrepancies between the two methodologies were apparent for releases consisting primarily of longer-lived radionuclides (e.g., ground irradiation pathway for FPFRTs, FEBTs, and FECF releases) or a specific radionuclide (e.g., IET-8). Because of this, it is of interest to further investigate differences between the two methodologies for specific exposure pathways as well as specific radionuclides.

For this more detailed comparison, we developed a release scenario to consist of all 98 radionuclides determined in the Selection of Important Radionuclides section to be most important to consider. In addition, certain Group 3, 4, and 5 releases involve some radionuclides that are not fission products and were not important to consider for the Group 1 and 2 releases. These radionuclides were also incorporated into the comparison. The release scenario consisted of 1 Ci of each radionuclide listed in Table B1.

Table B1. Radionuclides Considered for NCRP and RSAC Comparison Calculations

Ar- 41	Co- 60	I-132	La-142	Nd-149	Rb- 90	Sb-131	Te-129	Xe-131m	Y- 95
Ba-137m	Cr- 51	I-133	La-143	Pm-147	Rb- 90m	Sn-123	Te-129m	Xe-133	Zr- 95
Ba-139	Cs-134	I-134	Mn- 54	Pm-149	Rh-106	Sn-125	Te-131	Xe-135	Zr- 97
Ba-140	Cs-136	I-135	Mo- 93	Pm-151	Ru-103	Sr- 89	Te-131m	Xe-135m	
Ba-141	Cs-137	K- 42	Mo- 99	Pr-143	Ru-105	Sr- 90	Te-132	Xe-137	
Ba-142	Cs-138	Kr- 85	Mo-101	Pr-144	Ru-106	Sr- 91	Te-133	Xe-138	
Br- 84	Cs-139	Kr- 85m	Mo-102	Pr-145	S- 35	Sr- 92	Te-133m	Y- 90	
Cd-115m	Eu-155	Kr- 87	Na- 24	Pr-146	Sb-125	Sr- 93	Te-134	Y- 91	
Ce-141	Eu-156	Kr- 88	Nb- 95	Pr-147	Sb-126	Tc-101	U-234	Y- 91m	
Ce-143	Fe- 59	Kr- 89	Nb- 96	Pu-239	Sb-128m	Tc-102	U-235	Y- 92	
Ce-144	I-129	La-140	Nb- 97	Rb- 88	Sb-129	Tc-104	U-238	Y- 93	
Ce-146	I-131	La-141	Nd-147	Rb- 89	Sb-130	Te-127m	Xe-129m	Y- 94	

We ran RSAC primarily using the default parameters, particularly those that control distribution of radionuclides through the food chain. We did, however, attempt to select as many values as possible to correspond with the assumptions made for the NCRP screening factors (Table B2) to help isolate and identify the specific causes of differences between the two methodologies.

There are a number of differences in parameter values assumed by the RSAC program and the NCRP screening factors that we have not attempted to correlate. The RSAC program does not incorporate a soil ingestion exposure pathway, whereas the NCRP ingestion screening factors do, which results in a higher NCRP ranking value in some cases. RSAC assumes a physical weathering half-life of 0.0021 h^{-1}, while NCRP assumes a value of 0.003 h^{-1}, which would tend to result in lower NCRP values for the ingestion pathway. The RSAC-assumed harvest duration time period is set to 7 days for all

pathways, while the NCRP methodology assumes a period of 1 day for vegetables, 2 days for milk, and 7 days for meat. Similarly, the RSAC-assumed crop exposure time is 30 days for all pathways, while the NCRP methodology assumes a period of 60 days for vegetables and 30 days for both milk and meat (Table B2). There are a number of other differences between the two methodologies that may impact the calculated values, including variations in the element specific transfer factors, dose conversion factors, assumed chemical form of each radionuclide, and treatment of the specific decay schemes for each radionuclide.

Table B2. Assumptions for NCRP and RSAC Comparison Calculations

Assumption	RSAC	NCRP
Crop exposure time (days)	30[a]	60 (vegetables), 30 (meat), 30 (milk)[b]
Harvest duration (days)	7	1 (vegetables), 2 (meat), 7 (milk)[b]
Transit time (seconds)	500	500[c]
Buildup time in soil (years)	30	30[c]
Breathing rate	0.000254 (m^3 s^{-1})	8000 (m^3 y^{-1})[d]
Deposition velocity[e]	0.01157 m s^{-1}	1000 m d^{-1}[b]

[a] The default fraction of contaminated vegetables and forage for exposure times between 30 and <60 days is twice that for exposure times between 1 hour and <30 days. As a result, the ingestion value calculated by RSAC is effectively doubled when the exposure time is increased from 29.9 to 30 days.
[b] Source: Table 5.1 in NCRP (1996).
[c] Source: Table B.1 in NCRP (1996).
[d] Source: Table 7.1 in NCRP (1996).
[e] Value assumed for all radionuclides except noble gases where the deposition velocity is assumed to be 0.

We calculated NCRP and RSAC values as described for the Group 1 and 2 analyses and then computed ratios (NCRP value divided by RSAC value) for each radionuclide and each pathway. Figures B1 through B5 show the distribution of these ratios for each individual exposure pathway as well as the total or combination of all pathways. Ratios calculated for individual radionuclides are plotted in each histogram.

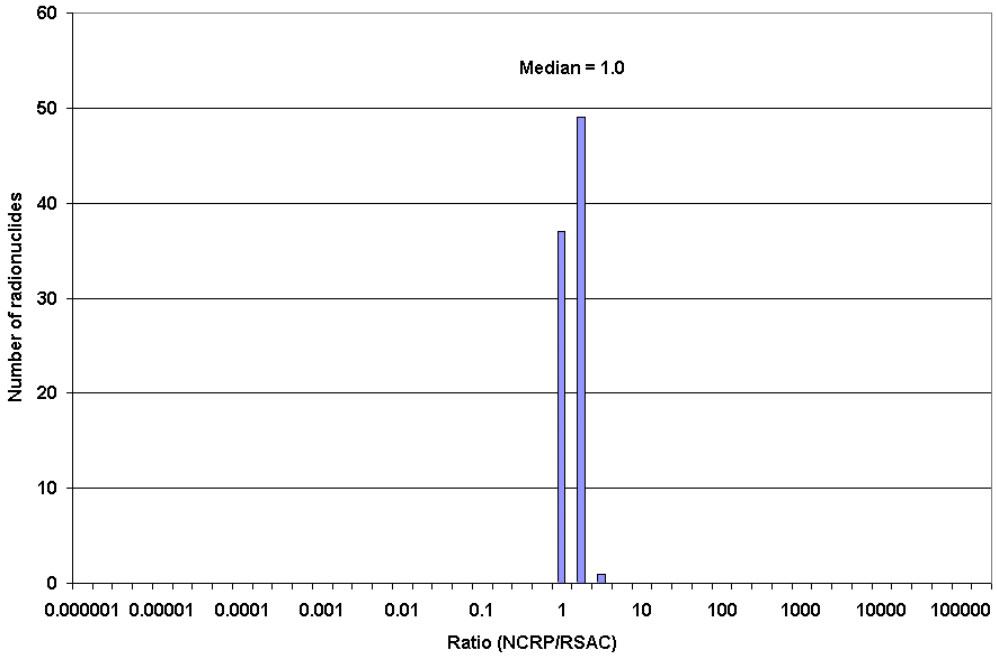

Figure B1. Histogram showing the distribution of ratios calculated for individual radionuclides (NCRP value divided by RSAC value) for the inhalation pathway. Radionuclides considered for this analysis are shown in Table B1.

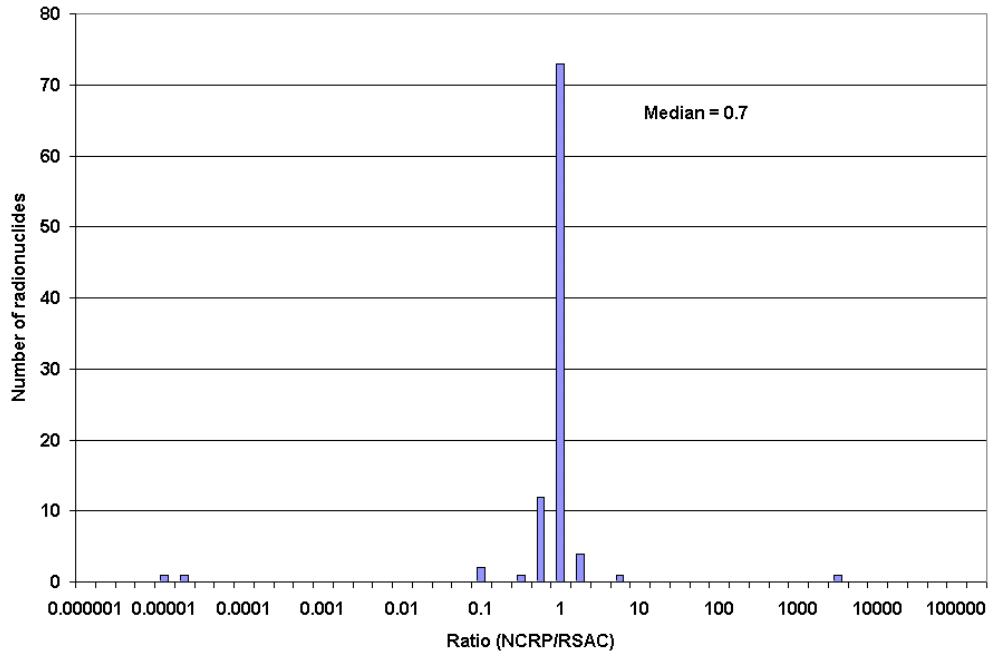

Figure B2. Histogram showing the distribution of ratios calculated for individual radionuclides (NCRP value divided by RSAC value) for the immersion pathway. Radionuclides considered for this analysis are shown in Table B1.

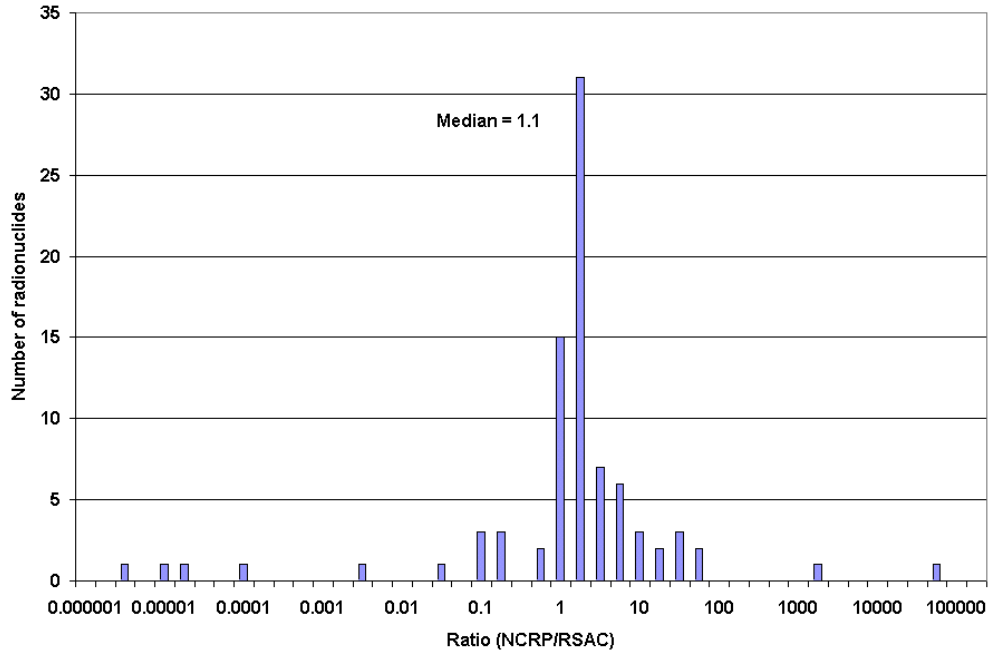

Figure B3. Histogram showing the distribution of ratios calculated for individual radionuclides (NCRP value divided by RSAC value) for the ground irradiation pathway. Radionuclides considered for this analysis are shown in Table B1.

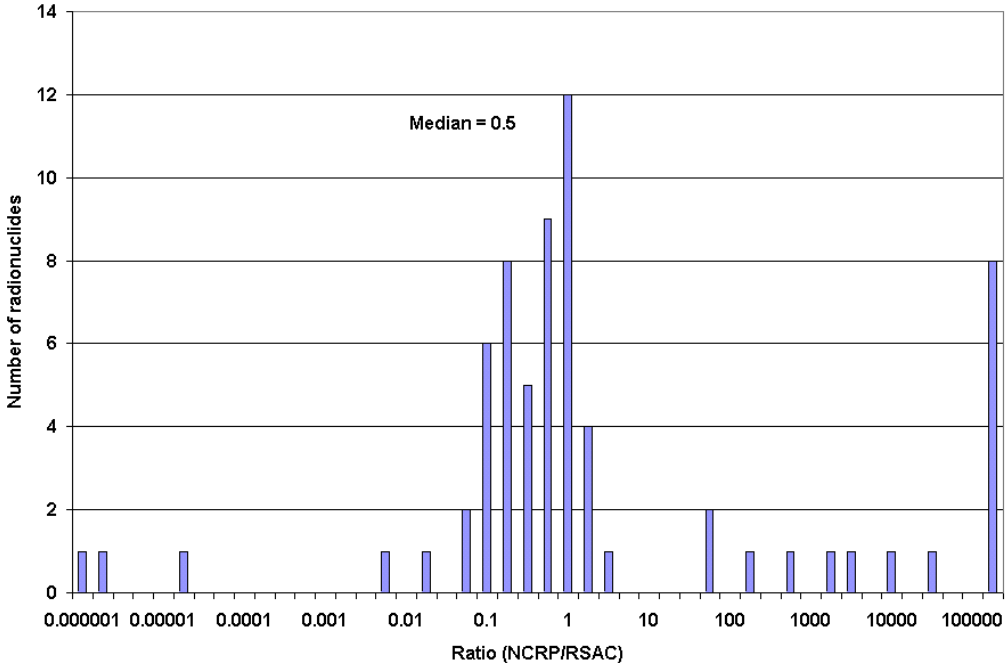

Figure B4. Histogram showing the distribution of ratios calculated for individual radionuclides (NCRP value divided by RSAC value) for the ingestion pathway. Radionuclides considered for this analysis are shown in Table B1.

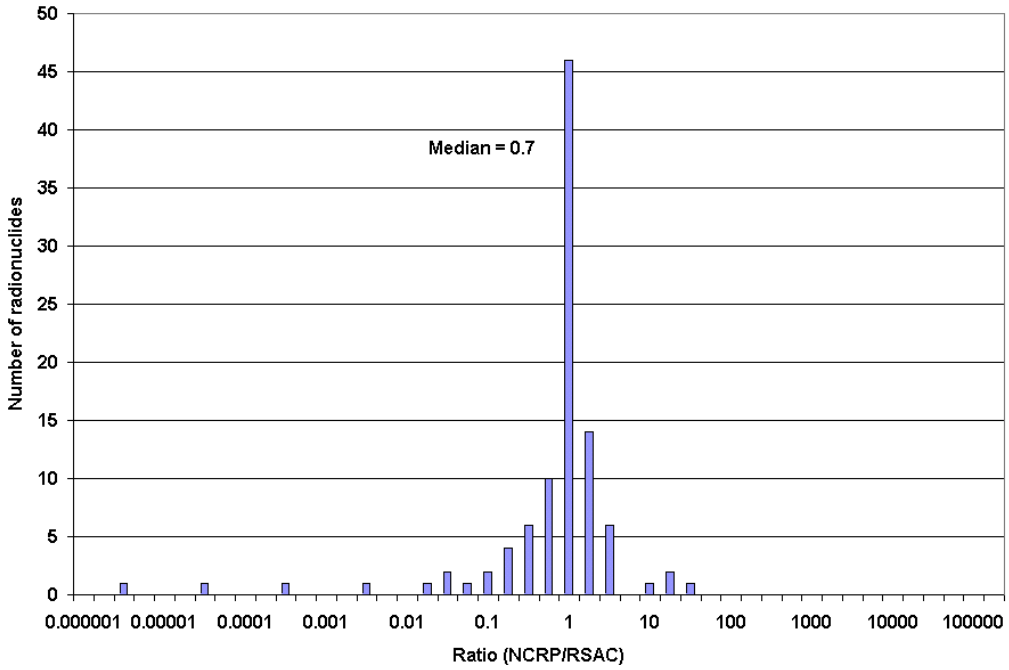

Figure B5. Histogram showing the distribution of ratios calculated for individual radionuclides (NCRP value divided by RSAC value) for all exposure pathways combined. Radionuclides considered for this analysis are shown in Table B1.

It is apparent that agreement between the NCRP and RSAC methodologies was strongest for the inhalation and immersion exposure pathways (Figures B1 and B2). This was not unexpected because the dose delivered through these pathways is independent of release duration, and the chronic release-specific assumptions inherent in the NCRP screening factors have little impact on the calculation of dose. There was also generally good agreement for the ground irradiation and ingestion pathways (Figures B3 and B4), although the greater number of assumptions and calculations to derive doses for these pathways resulted in a larger number of radionuclides showing significant differences between the two methodologies. More consistent agreement between the NCRP and RSAC methodologies was achieved when all pathways were considered than when only the ground irradiation or ingestion pathway was considered (Figure B5). The median ratio value for each pathway was generally consistent with the Group 1 and 2 ratios shown in Tables 19 and 28.

While ratios deviating significantly from 1 suggest either an underprediction or overprediction by NCRP when compared to RSAC, they provide no information about which methodology may be more correct. These high or low ratios simply indicate that two methodologies result in different values in some cases. The general agreement between the two methodologies suggests that the fact the NCRP air screening factors were designed to assess routine or long-term releases does not preclude their use for understanding the impact of short duration releases.

Because there are some significant discrepancies between the two methodologies for certain radionuclides and particularly the ground irradiation and ingestion exposure

pathways, though, we investigated further each radionuclide where the NCRP to RSAC ratio was less than 0.1 for any exposure pathway, indicating a lower NCRP value (i.e., potential underprediction) by greater than an order of magnitude (Table B3). To eliminate the potential for any influence by a parent to those radionuclides listed in Table B3, we calculated another set of RSAC values including only those radionuclides listed in Table B3 instead of the entire set of radionuclides listed in Table B1 as was done initially. The ratios presented in Table B3 are noted as "First" for the initial evaluation, which included all radionuclides in Table B1, and are noted as "Second" for the subsequent evaluation, which included only those radionuclides listed in Table B3.

Based on the resulting ratios from these two evaluations, which differed only in the set of radionuclides included in the release scenario, it is clear that in many cases, the potential underprediction by the NCRP methodology resulted from the treatment by RSAC of daughter product contributions resulting from decay of the parent subsequent to the time at which the plume reaches the exposure location. For example, the substantial underprection suggested for ^{137m}Ba did not result when the parent ^{137}Cs was not included; likewise, the substantial underprediction suggested for ^{144}Pr did not result when the parent ^{144}Ce was not included.

There were also instances where the potential underprediction by the NCRP methodology cannot be explained through the treatment by RSAC of daughter product contributions. For example, the significant potential underprediction suggested for ^{143}Pr and ^{106}Rh for the immersion pathway and for ^{143}Pr for the ground irradiation pathway was not appreciably changed by including only those radionuclides listed in Table B3 in the analysis. This result was not necessarily surprising for ^{143}Pr because ^{143}Pr is produced by a number of mechanisms, a principle one of which is as a direct fission product. It was not clear why the ratio for ^{106}Rh was not changed because it is produced primarily through decay of the parent ^{106}Ru, although the "Total" ratio is significantly changed when the parent radionuclide was not included. Although there are significant differences for these two radionuclides for certain pathways, these pathways represent a relatively insignificant contribution to the total dose through all pathways from these two radionuclides.

Another explanation for discrepancies between the two methodologies could be that the dose conversion factors used for ^{106}Rh for the immersion pathway and the dose conversion factors used for ^{143}Pr for both the immersion and ground irradiation pathways differ significantly between the two methodologies. Although we were not able to complete a comparison of assumed dose conversion factors as part of this project, it was apparent that there are some differences between the two methodologies. For example, the ratio of 0 for both ^{90}Sr and ^{90}Y for the immersion and ground irradiation pathways resulted from the fact that the dose factors used by the NCRP methodology assume no dose for either of these radionuclides through either of these pathways, and the dose factors used by the RSAC methodology do assume some dose through these two pathways. A more detailed investigation in this regard would require careful comparison of the dose factors assumed by the two methodologies, which are explicitly referenced in NCRP (1996) and Wenzel and Schrader (2001).

Table B3. NCRP to RSAC Ratios for Selected Radionuclides Whose Importance is Potentially Underestimated by the NCRP Methodology by Greater than One Order of Magnitude[a]

	Immersion		Ground irradiation		Ingestion		Total	
Radionuclide	First[b]	Second[c]	First	Second	First	Second	First	Second
Ba-137m	8.1E-02	8.1E-02	7.5E-06	1.1E+00			2.4E-05	1.2E-01
Eu-155					7.2E-02	7.2E-02		
I-132			3.1E-02	1.1E+00	1.6E-05	6.5E+02	3.1E-02	1.0E+00
La-140					7.1E-02	6.5E-01	9.7E-02	8.3E-01
Mn- 54					9.0E-02	9.0E-02		
Nb- 95					4.4E-03	1.1E-02	1.9E-02	5.0E-02
Nb- 97			7.1E-02	1.1E+00				
Pm-147					8.1E-02	8.1E-02		
Pr-143	1.5E-05	1.5E-05	1.2E-05	1.3E-05				
Pr-144			5.9E-05	8.3E-01	1.3E-07	nv[d]	2.5E-04	6.2E-01
Rb- 88			9.9E-02	1.0E+00				
Rh-106	6.7E-06	6.8E-06	2.0E-06	1.1E+00			2.0E-06	7.6E-03
Sb-125					9.7E-02	9.7E-02		
Sn-123					4.1E-02	4.1E-02	4.2E-02	4.2E-02
Sr- 89	6.8E-02	6.8E-02	6.1E-02	6.1E-02				
Sr- 90	nv[e]		nv[e]					
Te-127m					7.6E-02	7.6E-02	8.2E-02	8.2E-02
Te-129			2.2E-03	9.7E-01	1.1E-06	3.2E+10	2.1E-03	9.4E-01
U-234					3.7E-02	3.7E-02		
Y- 90	nv[e]		nv[e]		1.1E-02	7.6E-01	1.2E-02	7.9E-01

[a] There were no occurrences of this for the inhalation pathway; values are reported here only for instances where the ratio was less than 0.1.

[b] First refers to the evaluation that considered all of the radionuclides listed in Table B1.

[c] Second refers to the evaluation that considered only those radionuclides listed in this table.

[d] No value was calculated by RSAC, presumably because the majority of the ingestion dose for ^{144}Pr is through the ingestion of soil, which is not considered by RSAC.

[e] No value was calculated using the NCRP methodology because no dose contribution is assumed for this pathway.

We also investigated radionuclides where the NCRP to RSAC ratio was greater than 10 for any exposure pathway, indicating a higher NCRP value (i.e., potential overprediction) by more than an order of magnitude (Table B4). To eliminate the contribution by daughter products, we calculated another set of NCRP values using screening factors that excluded any contribution by daughter products. The ratios presented in Table B4 are noted as "First" for the initial evaluation, which utilized the NCRP screening factors that include daughter product contributions, and are noted as "Second" for the subsequent evaluation, which utilized the NCRP screening factors excluding any daughter product contributions.

Based on the resulting ratios from these two evaluations, it is clear that in many cases, the potential overprediction by the NCRP methodology results from the treatment by NCRP of daughter product contributions resulting from decay of the parent during a period of 30 years with a constant annual average air concentration. For example, it is evident that a significant part of the NCRP ingestion pathway value for such radionuclides as ^{141}La, ^{149}Nd, ^{131m}Te, and ^{133m}Te results from daughter product contributions. For ^{137}Cs, the immersion and ground irradiation pathway values are derived entirely from daughter product ^{137m}Ba contributions. Additionally, in many instances (noted in Table B4) the high ratio is influenced substantially by inclusion of soil ingestion into the ingestion pathway by the NCRP methodology, which is not incorporated into the RSAC methodology. It is also possible that much of the potential overprediction results from the NCRP assumption of deposition and subsequent buildup in the soil over a period of 30 years with a constant annual air concentration.

Another source of potential overprediction for some radionuclides could be related to the assumed chemical form. NCRP (1996) notes that "…the chemical form yielding the highest effective dose was selected…In the extreme, the approach could introduce a conservatism in excess of a factor of 10." The specific assumptions made by RSAC regarding chemical form are not entirely clear, although Wenzel and Schrader (2001) notes that "Correction is made for the chemical state of each radionuclide according to the ICRP-30 designated clearance classes of D, W and Y as shown in Table A3 for the clearance pathways shown in Figure A-3." Again, we were not able to investigate in detail the specific assumptions made by each methodology.

NCRP (1996) also notes that the concentration ratios for transfer from soil to vegetables and forage for cesium were adjusted upward to avoid underestimating the uptake in special situations such as those existing for the nutrient-poor sandy soils of the southeast coastal plain of the United States. Because of this ajustment, the exposure to cesium isotopes via the ingestion of contaminated foodstuffs should result in a potentially large overestimate. The transfer factors assumed by NCRP for cesium are clearly higher than those assumed by RSAC; however, such an overestimate is not apparent based on the NCRP to RSAC ratios calculated for ^{137}Cs and ^{136}Cs, which are 0.3 and 1.1, respectively. Again, we were not able to investigate in detail the specific assumptions made by each methodology that govern uptake and transfer through the food chain.

Table B4. NCRP to RSAC Ratios for Selected Radionuclides Whose Importance is Potentially Overestimated by the NCRP Methodology by Greater than One Order of Magnitude[a]

	Immersion		Ground irradiation		Ingestion		Total	
Radionuclide	First[b]	Second[c]	First	Second	First	Second	First	Second
Ba-139					1.6E+08	1.6E+08[d]		
Ba-140			1.4E+01	1.2E+00				
Cs-137	2.4E+03	0.0E+00	4.0E+04	0.0E+00				
I-129			2.2E+01	2.2E+01				
La-141					3.8E+05	6.4E+03		
La-142					3.3E+08	3.3E+08[d]		
Mo- 93			3.2E+01	3.0E+01				
Nb- 97					8.6E+06	8.6E+06[d]		
Nd-149					1.5E+11	3.0E+07[d]		
Pr-145					1.5E+03	1.5E+03		
Pu-239			2.7E+01	2.7E+01				
Ru-105					6.0E+06	1.3E+05		
Sb-129					2.9E+04	6.8E+02		
Sb-130					7.9E+07	7.9E+07[d]		
Sb-131							1.3E+01	7.3E-01
Sr- 91					4.1E+01	6.9E+00		
Sr- 92					3.1E+03	8.4E+02		
Te-131							1.1E+01	3.1E-01
Te-131m					3.8E+01	8.7E-01	2.3E+01	9.8E-01
Te-132			1.1E+01	1.1E+00				
Te-133m					9.7E+17	7.2E+13[d]		
U-234			2.9E+01	2.9E+01				
U-235			3.5E+01	3.1E+01				
U-238			1.4E+03	3.1E+01				
Y- 92					1.0E+04	1.0E+04		
Y- 93					4.5E+02	4.5E+02		
Zr- 97					1.7E+02	1.7E+02		

[a] There were no occurrences of this for the inhalation pathway; values are reported here only for instances where the ratio was less than 0.1

[b] First refers to the evaluation that included the NCRP screening factor daughter product contribution

[c] Second refers to the evaluation that excluded the NCRP screening factor daughter product contribution

[d] High ratio is influenced substantially by inclusion of soil ingestion into the ingestion pathway by the NCRP methodology, which is not incorporated into the RSAC methodology

It is important to note that the discrepancies, in either direction, between the NCRP and RSAC methodologies when all pathways are considered (i.e., "Total") are relatively small, as evidenced by the data presented in Figure B5 and the ratios presented in the "Total" columns of both Tables B3 and B4. This consistent agreement between the two

methodologies suggests that significant discrepancies between the two methodologies are generally confined to exposure pathways that are not major contributors to the sum of all pathways. The consistent agreement between ranking values estimated by the two methodologies for the Group 1 and Group 2 release events, most of which were comprised of a number of different radionuclides, suggests that significant discrepancies are also primarily confined to radionuclides that are not major contributors to the overall importance of a release that includes a mixture of fission products.

The close agreement between the ranking orders developed using both methodologies for the Group 1 and Group 2 release events demonstrates that the NCRP screening factors are suitable for understanding the relative importance of episodic or short-term releases. Furthermore, the general agreement between the NCRP and RSAC methodologies suggests that the NCRP screening factors may also be suitable in many cases for estimating potential doses for short-duration releases. This is particularly true for specific radionuclides, the inhalation and immersion pathways, and release events consisting of a broad mix of different fission products resulting from reactor operations or criticality events. However, it is clear that caution should be exercised because of the large differences seen between the NCRP and RSAC methodologies for certain radionuclides.

Further elucidation of the specific causes for the discrepancies noted between the NCRP and RSAC methodologies would require a detailed investigation into a number of parameters and equations, including the impact of variations in element-specific transfer factors, dose conversion factors, assumed chemical form of each radionuclide, and treatment of the specific decay schemes for each radionuclide.

Because some releases are appropriate to evaluate excluding the ingestion pathway contribution, we also examined the ratios resulting from comparing NCRP and RSAC values for each radionuclide that incorporate the inhalation, immersion, and ground irradiation pathways only. No additional radionuclides emerged with ratios less than 0.1. Three additional radionuclides emerged with ratios greater than 10 (^{93}Mo, ^{140}Ba, and ^{137}Cs). This observation is consistent with the ground irradiation pathway ratios calculated for the Group 1 IET-8 release (Table 19), which consisted primarily of ^{93}Mo, and the Group 2 FPFRT releases consisting primarily of long-lived radionuclides, like ^{137}Cs (Table 28). With the exception of ^{137}Cs, none of the Group 3, 4, or 5 releases consists soley or primarily of these additional radionuclides.

All Group 3, Group 4, and Group 5 evaluations are based only on the NCRP screening factors. None of the releases consists soley or primarily of any of the four radionuclides in Table B3 with "Total" ratios less than 0.1 when evaluated without the presence of any parent radionuclides (^{95}Nb, ^{106}Rh, ^{123}Sn, and ^{127m}Te). As a result, the Group 3, 4, and 5 evaluations based only on the NCRP screening would not be expected to underestimate any ranking value by greater than one order of magnitude by comparison to an RSAC-calculated ranking value. In addition, none of the releases consists soley or primarily of any of the three radionuclides in Table B4 with "Total" ratios greater than 10 (^{131}Sb, ^{131}Te, and ^{131m}Te). As a result, the Group 3, 4, and 5 evaluations based only on the NCRP screening would not be expected to overestimate any "Total" ranking value by greater than one order of magnitude by comparison to an RSAC-calculated ranking value.

Based on this sum of evidence, there is no indication that using the NCRP air screening factors to generate ranking values for the Group 3, 4, and 5 releases would result

in any significant errors in the ranking order. Therefore, using the NCRP methodology to calculate relative ranking values for the Group 3, 4, and 5 releases is considered appropriate. However, because of the potentially significant overestimate that is possible for releases consisting soley or primarily of ^{137}Cs when the ingestion pathway is not considered, this is noted for releases appearing as important within a given group.